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FAST NEUTRON GAUGING FEASIBILITY STUDY

Final Report

R. J. Schofield, Project Leader

Project Team:

J. S. Carson

R. V. Heckman

Published March 1978

Prepared for the United States Department of Energy
Under Contract Number EY-76-C-04-0613.

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**Kansas City
Division**

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Project Leader:
R. J. Scholfield

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J. S. Carson
✓ R. V. Heckman — see pg. 2

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FAST NEUTRON GAUGING FEASIBILITY STUDY

BDX-613-1770 (Rev.), UNCLASSIFIED Final Report, Published March 1978

✓
Prepared by R. V. Heckman, D/442

A study was conducted to determine the ability of a collimated neutron beam to locate defects in polyurethane foam encapsulated between sintered-iron and stainless-steel jackets. Test results indicate that although a californium-252 source can be used, an iron-filtered neutron beam provides the greatest sensitivity. Voids having a diameter of 0.2 cm, an effective area of 1 cm², and a volume of 0.14 cm³ were successfully detected.

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The Bendix Corporation
Kansas City Division
P. O. Box 1159
Kansas City, Missouri 64141

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SUMMARY

An investigation was made of the feasibility of using a collimated neutron beam for locating voids and other internal defects in the polyurethane-foam outer jacket of an assembly. This foam is located between steel plates which prevent the use of conventional gamma-gauging methods. The study was conducted at the University of Missouri Research Reactor (MURR) facility located at Columbia, Missouri.

Although the test results indicated that a californium-252 neutron source could be used for detecting voids in the polyurethane, an iron-filtered neutron beam was found to provide the greatest sensitivity. Voids having a diameter of 0.2 cm, an effective area of 1 cm², and a volume of 0.14 cm³ were successfully detected.

The ¹⁰B_F₃-filled detector was found to be the most sensitive. By increasing the number of detectors used, the time required to scan the entire enclosure can be reduced to a reasonable value.

DISCUSSION

SCOPE AND PURPOSE

This project was initiated to determine the sensitivity of various neutron-gauging techniques to polyurethane foam. Specifically, it sought to determine the smallest void or other defect that could be located through the use of each technique and, on that basis, to determine the most sensitive technique for production use.

In all, a series of seven different configurations (three for the bismuth filter and four for the iron filter) were attempted using three different thermal detectors (^3He , $^{10}\text{BF}_3$, and LiI), as well as a proton-recoil detector (H_2) which is effective for detecting fast neutrons. The seven configurations are shown in Figures 1 and 2.

PRIOR WORK

In the past, Bendix Kansas City has used neutron radiography to examine electronic components and other assemblies; however, the foam part is the first application of a nonfilm neutron-gauging technique.

ACTIVITY

The construction of the part prevents the use of conventional gamma-gauging methods, since the steel inner and outer linings make impossible the inspection of the lower-density polyurethane foam that is sandwiched in between. (The polyurethane foam has a density of approximately 480.55 kg/m^3 , or 30 lb/ft^3 .) A literature search of nondestructive inspection methods indicated that neutron-gauging or ultrasonic techniques were the only methods applicable. Neutrons have the unique capability of penetrating steel while being highly absorbed by such organic materials as polyurethane.

The absorption spectrum of iron (Fe), with regard to neutrons, exhibits a window in the 24-keV region. Thus when a neutron beam impinges on Fe , an almost-monoenergetic beam of 24 keV results. This physical property was thought to be advantageous for this particular application, since 24-keV neutrons have a high cross section with regard to the hydrogen that is contained in the foam material.

Another promising approach consisted of using a californium-252 (^{252}Cf) source instead of a reactor beam for testing. This method would allow on-site testing, and it might be more economical than

Figure 1. Neutron-Gauging Configurations Using Bismuth Filter

using a reactor for the neutron source. The suggestion was made that the use of a bismuth filter would provide a neutron spectrum which would closely match the spectrum of a californium-252 source.

Because Bendix Kansas City does not have a neutron facility within the plant, a proposal was requested from the University of Missouri Research Reactor (MURR) for evaluating the feasibility of using a collimated neutron beam for detecting voids or other defects in polyurethane foam encapsulated between sintered-iron and

Figure 2. Neutron-Gauging Configurations Using Iron Filter

stainless-steel jackets. Upon receipt of the proposal, a contract was negotiated for MURR to perform the investigation. A complete report of the work that was done is included in the Appendix of this report.

ACCOMPLISHMENTS

Both iron- and bismuth-filtered (simulated californium 252) neutrons were found capable of adequately detecting voids in polyurethane foam. Of the two flux distributions, the iron-filtered beam provided the greatest sensitivity for detecting voids in polyurethane sandwiched between steel. This iron-filtered beam yields 24-keV neutrons which penetrate the steel and still provide good sensitivity for detecting variations in the foam material.

An expression was obtained for relating the neutron flux to the void size. Voids having a diameter of 0.2 cm, an effective area of 1 cm^2 , and a volume of 0.14 cm^3 were successfully detected.

The $^{10}\text{BF}_3$ -filled detector was found to be the most sensitive. By increasing the number of detectors used, the time required to scan the entire enclosure can be reduced to a reasonable value.

FUTURE WORK

If the decision is made, tests of this type will be required in the future. Neutron gauging will be considered as a possible production inspection method.

Appendix

MURR REPORT ON NEUTRON GAUGING

FAST NEUTRON GAUGING FEASIBILITY STUDY

by

Francis Y. Tsang

Don M. Alger

Robert M. Brugger

Prepared For

Bendix Corporation

Purchase Order: G 164876

Project Director: Robert Schofield

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University of Missouri

Research Reactor Facility

Columbia, Missouri 65201

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*Original page numbers have been changed for compatibility with this report.

I. SUMMARY

Neutron transmission measurements were run to determine what size voids can be detected in polyurethane sandwiched in between steel plates. The sensitivity of three types of thermal neutron detectors imbedded in a paraffin moderator were tested to determine the most efficient fast neutron counter. The $^{10}\text{BF}_3$ filled detector was found to be more sensitive than the ^3He and ^6LiI detectors for two flux distributions. Two neutron flux distributions, one a Bi filtered beam and the second an Fe filtered beam, were tested to determine which flux provided the better sensitivity to detect a void in polyurethane sandwiched in steel. The Fe filtered beam yields neutrons of 24 keV which see through the cross section window of iron. The Bi filtered beam is a flux distribution similar to that from a ^{252}Cf source. Both beams were able to detect voids. The Fe filtered beam is more sensitive for polyurethane of less than 10 cm thickness while the Bi filtered beam is more sensitive for greater than 10 cm. An expression to relate the required neutron flux to void sized was derived, and the relationship to collimation area and counting period were compared. The result of the Bi filter beam measurement indicated that ^{252}Cf can be used to detect voids of a certain size. Recommendations to increase void sensitivity and decrease scanning time are given.

II. PURPOSE

The purpose of this investigation is to evaluate fast neutrons as a method to detect voids in polyurethane located between steel plates. The transmission of neutrons through a sample of polyurethane with variable void sizes was measured for two fast neutron distributions.

The fast neutron fluxes were produced by neutron beam filters so that the more effective distribution could be selected. One filter produced neutrons of 24 keV which are highly penetrating for the steel plates, while the second, a Bi filter, produced a flux distribution similar to that of ^{252}Cf . Penetration of the fast neutrons through the sample slab were detected by three thermal detectors imbedded in a moderator. These three detectors were tested to determine the one with the highest counting rate and sensitivity.

III. EXPERIMENT

A versatile neutron beam facility has been installed at the 'F' beam port of MURR¹. With this facility, neutron fluxes of a predetermined energy distribution can be produced by selecting the filter material. The filter and the moderator that were used in this investigation are described in the following sections.

A. Neutron Filters

1. Bi Filter

The transmitted neutron spectrum through 19 cm of Bi was measured by Miller using a NE-213 scintillation detector. The measured spectrum is broad and has a most probable neutron energy at 1.2 MeV. This can be compared to a most probable energy of 0.75 MeV for a fission spectrum² and 0.65 MeV for a ²⁵²Cf spectrum³. Miller's comparison is shown in Figure 1. The more rapid drop off of the Bi spectrum at about 2 to 3 MeV corresponds to a resonance in the total cross section of Bi. The drop off in flux below 1 MeV represent the cut off of the detector and is not a true representation of the flux from the Bi filter below 1 MeV. The neutron flux from the filter is 2.0×10^6 n/cm²-sec 50 cm from the end face of the filter assembly. A ¹⁰B powder filter was placed in the beam to remove the thermal neutrons. The gamma intensity in the beam was measured to be about 2 R/hr. In spite of the difference noted above, it is estimated that the flux distribution from the Bi filter is a good approximation to

FIG 1

A comparison between the measured Bi filter spectrum to the calculated fission spectrum and measured ²⁵²Cf spectrum

that of ^{252}Cf .

2. Fe Filter

The Fe filter used in this investigation consisted of 50.8 cm of Fe, 20.3 cm of Al and 5.1 cm of S. The neutron flux from this assembly is 1.2×10^6 n/cm²-sec with a FWHM of ~ 2 keV. With the ^{10}B powder filter in the beam to remove the thermal neutrons, the background from neutrons of energies other than the 24 keV neutron group is $\sim 0.15\%$. The gamma intensity in the beam was measured to be < 44 mR/hr. A measured spectrum of the Fe filtered beam is shown in Figure 2.

B. Neutron Detectors

Three neutron detectors were imbedded in a neutron moderator. The neutron moderator used in this investigation consisted of a 15 cm diameter cylindrical paraffin block. With a 8 cm deep by 2.5 cm diameter hole on the front face of the block. The surfaces of the block were covered with Cd foils. Two through holes, that were parallel to the center hole, were used to imbed the $^{10}\text{BF}_3$ and ^3He detectors.

Another 5.0 cm deep hole for the ^6LiI detector was machined vertical to the center line of the neutron beam on the body of the paraffin block. Boronated polyethylene collimators were placed in front of the neutron detector to collimate the fast neutrons. A sketch of the neutron detector unit is shown in Figure 3.

FIG 2

Transmitted neutron spectrum of Fe filtered beam
measured by H_2 gas filled proportional counter

FIG 3

NEUTRON DETECTOR FOR NEUTRON GAUGING MEASUREMENT

NOT TO SCALE

IV. MEASUREMENT AND RESULT

To measure the transmission of the polyurethane block the following procedure was followed:

1) The spectrum from the Bi filter or Fe filter was used to produced fast neutrons for the transmission measurement.

2) A collimator was placed in between the end face of the filter and the sample to collimate the neutron beam to a definite area. The experimental set up is shown in Figure 4.

3) Three detectors, $^{10}\text{BF}_3$, ^3He , and ^6LiI were inserted into the moderator to detect the transmitted fast neutrons. The counting rates for each of the detectors were recorded for each of the two neutron flux distribution as a function of the thickness of polyurethane. The data are shown in Table 1 and 2. The results are reproduced in Figure 5 and 6 with the counting rate as a function of the total polyurethane thickness sandwiched in between 13 cm steel plates.

Fig 4

EXPERIMENTAL LAY OUT FOR FAST NEUTRON GAUGING

NOT TO SCALE

TABLE 1
Bi Filter
Detector Counts/Sec

Thickness of Polyurethane in cm.	$^{10}\text{BF}_3$	^3He	^6LiI
0.00	630 $\pm$ 9	556 $\pm$ 6	362 $\pm$ 4
0.64	581 $\pm$ 8	491 $\pm$ 5	322 $\pm$ 4
3.18	401 $\pm$ 7	352 $\pm$ 4	233 $\pm$ 6
4.29	342 $\pm$ 3	300 $\pm$ 4	196 $\pm$ 2
5.72	290 $\pm$ 5	250 $\pm$ 4	161 $\pm$ 3
7.62	224 $\pm$ 4	200 $\pm$ 2	127 $\pm$ 2
8.23	200 $\pm$ 3	180 $\pm$ 3	114 $\pm$ 1

Transmitted neutron count rate for various thicknesses of polyurethane. These measurements were taken with the polyurethane between 13 cm thick steel slabs. A 0.36 cm² collimator was used.

TABLE 2
Fe Filter
Detector Counts/Sec

Thickness of Polyurethane in cm.	$^{10}\text{BF}_3$	^3He	^6LiI
0.00	651 $\pm$ 9	514 $\pm$ 4	499 $\pm$ 5
0.64	484 $\pm$ 4	383 $\pm$ 4	379 $\pm$ 3
3.18	188 $\pm$ 1	150 $\pm$ 2	159 $\pm$ 3
4.29	122 $\pm$ 2	99 $\pm$ 2	110 $\pm$ 2
5.72	76 $\pm$ 1	64 $\pm$ 1	76 $\pm$ 2
7.62	42 $\pm$ 1	38 $\pm$ 1	50 $\pm$ 1
8.23	33 $\pm$ 0.5	31 $\pm$ 1	44 $\pm$ 1

Transmitted neutron count rate for various thicknesses of polyurethane. These measurements were taken with the polyurethane between 13 cm thick steel slabs. A 1.26 cm² collimator was used.

FIG 5

Transmitted neutron intensity with Bi filter as a function of polyurethane thickness sandwiched in between 13 cm of steel plates. A collimator of 0.36 cm^2 is used

FIG 6

Transmitted neutron intensity with Fe filter as a function of polyurethane thickness sandwiched in between 13 cm of steel plates. A collimator of 1.26 cm^2 was used

V. INTERPRETATION

The $^{10}\text{BF}_3$ detector imbedded in a paraffin moderator has the highest counting rates. The $^{10}\text{BF}_3$ detector is the most sensitive and was used for subsequent measurements. The absolute sensitivity for the $^{10}\text{BF}_3$ detector with the two flux distributions were determined by normalizing the counting rate to the incident flux and collimation area. The results as a function of polyurethane thickness for the two flux distributions are shown in Figure 7 and 8.

The comparison of the two figures shows that the Fe filter with a larger macroscopic cross section for hydrogen has a better sensitivity to void size than the Bi filter. This is due to the fact that the Fe filtered beam is 24 keV and these neutrons correspond to the neutron window in the total cross section of Fe. This effect provide more neutrons to interact with the hydrogen in the polyurethane for the Fe filter than the Bi filter since the 24 keV neutrons see through the cross section window in Fe.

FIG 7

Normalized values of counts/incident neutrons of the $^{10}\text{BF}_3$ detector with the Bi filter. A 0.36 cm^2 collimator was used

FIG 8

Normalized value of counts/incident neutron of the $^{10}\text{BF}_3$ detector with the Fe filter. A 1.26 cm^3 collimator was used

Due to the fact that, the Bi filtered beam has a flux distribution similar to a ^{252}Cf source, the discussion in the following will be based on the information provided by the Bi filter.

The relationship between a required neutron flux (ϕ) and a certain void thickness (ΔX) with 2σ uncertainty in counting statistics for a given neutron beam area and counting period is derived in the following manner.

The absolute sensitivity of the $^{10}\text{BF}_3$ detector is shown in the following sketch:

Thickness of polyurethane sandwiched
in between 13 cm of steel plates.

Where C_o = Counts/incident neutrons for zero thickness of polyurethane sandwiched in between 13 cm of steel plates.

T = Total thickness of polyurethane without void.

C = Counts/incident neutrons corresponding to the thickness of polyurethane without void.

Suppose the total yield/sec of incident neutrons in the investigation is to be N_o , so from the equation of transmission the counts through a thickness T can be determined by

$$C N_o = C_o N_o e^{-\Sigma T} \quad (1)$$

Where Σ = macroscopic cross section of the polyurethane.

$C = C_o e^{-\Sigma T}$ = counts at thickness T

and we can also obtain

$$\Sigma = \ln \left| \frac{(C)}{C_o} \right| / T \quad (2)$$

The uncertainty in the measured counts can be determined at the 95% confidence limit by determining twice the standard deviation. This is approximately $2(CN_o)^{1/2}$. Setting this

equal to the signal change, ΔC , from the above figure gives:

$$\Delta C = 2\sqrt{CN_0} \quad (3)$$

Let P equal the fractional uncertainty in counts, therefore:

$$P = \frac{2\sqrt{CN_0}}{CN_0} = \frac{2}{\sqrt{CN_0}} = \frac{4}{\Delta C} \quad (4)$$

The thickness corresponding to ΔC counts is ΔX which can be written as:

$$\Delta X = T - X$$

Where X is the actual equivalent polyurethane thickness or

$$X = T - \Delta X \quad (5)$$

using equation (1) the difference in counts between thickness T and X can be expressed as

$$C_0 N_0 (e^{-\Sigma(T-\Delta X)} - e^{-\Sigma T}) = \Delta C \quad (6)$$

So substituting (5) into (6) we obtain

$$C_0 N_0 (e^{-\Sigma(T-\Delta X)} - e^{-\Sigma T}) = \Delta C$$

or

$$\Delta C = C_0 N_0 e^{-\Sigma T} (e^{\Sigma \Delta X} - 1) \quad (7)$$

rearranging the equation we obtain

$$\begin{aligned} N_0 &= \frac{\Delta C}{C_0 e^{-\Sigma T} (e^{\Sigma \Delta X} - 1)} \\ &= \frac{4}{C_0 e^{-\Sigma T} P (e^{\Sigma \Delta X} - 1)} \end{aligned} \quad (8)$$

Therefore an expression to relate the thickness T and X to P is

$$\begin{aligned} P &= \frac{C_0 N_0 e^{-\Sigma(T-\Delta X)} - C_0 N_0 e^{-\Sigma T}}{C_0 N_0 e^{-\Sigma T}} \\ &= \frac{e^{-\Sigma T + \Sigma \Delta X} - e^{-\Sigma T}}{e^{-\Sigma T}} \\ &= (e^{\Sigma \Delta X} - 1) \end{aligned}$$

So equation (8) can be expressed as

$$N_0 = \frac{4}{C_0 e^{-\Sigma T} (e^{\Sigma \Delta X} - 1)^2}$$

Since the neutron flux $\phi = \frac{N_0}{At}$

Where A = neutron beam area
t = time in sec.

Therefore

$$\phi = \frac{4}{C_0 A t e^{-\Sigma T} (e^{\Sigma \Delta X} - 1)^2} \quad (9)$$

Equation (9) shows that the neutron flux needed for an investigation is inversely proportional to the area of collimation, the counting period and the thickness of the void under investigation. For example, if the void size is 0.5 cm deep and the area of collimation is 1 cm², the neutron flux required in the investigation with the counting period of 10 sec can be obtained by equation (9).

$$\phi = \frac{4}{C_0 A t e^{-\Sigma T} (e^{\Sigma \Delta X} - 1)^2}$$

and $C_0 = 1.63 \times 10^{-4}$ counts/incident neutron with the Bi filter.

$$A = 1 \text{ cm}^2$$

$$t = 10 \text{ sec}$$

$$\Sigma = 0.1376 \text{ cm}^{-1} \text{ derived from the transmission curve from Bi filter in Figure 7.}$$

$$T = 8.33 = \text{total thickness of polyurethane sandwiched in between 13 cm of steel plates.}$$

$$\Delta X = 0.5 \text{ cm}$$

$$\begin{aligned} \text{So } \phi &= \frac{4}{1.63 \times 10^{-4} \times 1 \times 10 \times e^{-0.1376 \times 8.33} (e^{0.1376 \times 0.5} - 1)^2} \\ &= 1.52 \times 10^6 \frac{\text{n}}{\text{cm}^2\text{-sec}} \end{aligned}$$

If for example a ²⁵²Cf is employed in the investigation and is placed $r = 25$ cm from the inner surface of the barrel as in the following sketch:

Cross Section sketch of a steel barrel
with a ^{252}Cf source (not to scale)

The strength of the ^{252}Cf source can be determined as the following:

$$\frac{\phi_{^{252}\text{Cf}}}{4\pi r^2} = 1.52 \times 10^6 \frac{\text{n}}{\text{cm}^2\text{-sec}}$$

Strength of ^{252}Cf at the center of the barrel is

$$\begin{aligned}\phi_{^{252}\text{Cf}} &= 4\pi \times 25^2 \times 1.52 \times 10^6 \frac{\text{n}}{\text{cm}^2\text{-sec}} \times \text{cm}^2 \\ &= 1.19 \times 10^{10} \frac{\text{n}}{\text{sec}}\end{aligned}$$

which corresponds to a 6.0 mg ^{252}Cf source.

The magnitude of source strength can be reduced and the sensitivity to smaller voids can be increased if the value of C_0 of the detector can be increased.

This can be done by increasing the total number of $^{10}\text{BF}_3$ detectors in the moderator to 8. With this modification C_0 with 8 $^{10}\text{BF}_3$ detectors will be equal to 1.30×10^{-3} counts/incident neutron, which in turn will reduce the ^{252}Cf to a $1.49 \times 10^9 \frac{n}{\text{sec}}$ source. This corresponds to a 0.75 mg ^{252}Cf .

If the smallest void to be detected is 0.2 cm with an area of 1 cm^2 and the value of C_0 is modified to 1.30×10^{-3} by increasing the number of $^{10}\text{BF}_3$ detector to 8. The source strength required would be:

$$4 \times \pi \times 25^2 \times \frac{4}{1.63 \times 10^{-4} \times 8 \times 10 \times e^{-0.1376 \times 8.33} (e^{0.1376 \times 0.2} - 1)^2}$$

$$= 9.72 \times 10^9 n/\text{sec.} \quad \text{This corresponds to 4.8 mg of } ^{252}\text{Cf}.$$

If a 10 mg ^{252}Cf is employed in the investigation the parameters in the investigation are summarized in the following table.

$$10 \text{ mg } ^{252}\text{Cf} = 2 \times 10^{10} \frac{n}{\text{sec}} \quad r = 25 \text{ cm with the source at the center}$$

$$\Sigma = 0.1376 \text{ cm}^{-1} \quad C_0 = 1.3 \times 10^{-3} \text{ with 8 } ^{10}\text{BF}_3 \text{ detectors}$$

$A(\text{cm}^2)$	$T(\text{sec})$	$\Delta X(\text{cm})$	Volume of void $V(\text{cm}^3)$
1	10	0.14	0.14
2	10	0.10	0.20
3	10	0.08	0.24
4	10	0.07	0.28
5	10	0.06	0.32
1	5	0.19	0.19
2	5	0.14	0.28
3	5	0.11	0.34
4	20	0.05	0.20

The results obtained from the above table demonstrate that the counting period chosen is inversely proportional to the void size and collimation area of the neutron beam.

The total scanning time for a 30 cm radius and 100 cm height barrel depends on the total collimation area. The total area of collimation under investigation can be increased by increasing the total number of detectors around the outer surface of the barrel as shown in the sketch.

A sketch to demonstrate the possible set up of the moderators around the steel barrels (not to scale)

The surface area of the barrel in the above description is 18550 cm². If the counting period is 10 sec to investigate an area of 1 cm² for a void volume toleration of 0.14 cm³, then the total scanning time will be 52 hours with one moderator set up. The scanning time can be reduced by a factor of 6 if the total number of moderators on the outer surface of the barrel is increased to the maximum permissible number as in the previous sketch.

VI. RECOMMENDATION

- 1) The total number of detectors imbedded in the paraffin moderator should be increased to the maximum number to improve the absolute sensitivity to void sizes or reduce scanning time.
- 2) The ^{252}Cf neutron source should be extended to as high a permissible mass as possible without causing a health hazard to the investigators. (A 10 mg of ^{252}Cf is a reasonable choice.)
- 3) The area of collimation should be expanded to a maximum corresponding to the void size sensitivity required.
- 4) The total number of detectors around the outer surface of the barrel should be increased to reduce the scanning time for the entire inspection.
- 5) If a 10 mg ^{252}Cf source is not adequate to locate the size of void desired in the time desired, consideration should be given to taking the barrels to the Weapon Neutron Research Facility (WNRF) at LASL where a neutron source of $>10^{14} \frac{n}{\text{sec}}$ will be available. Since this source is 1000 times more intense than a ^{252}Cf source, smaller voids could be observed or the counting time could be greatly reduced.

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SPECIAL PROJECTS: Nondestructive Testing

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