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ENGINEERING THE SLOWPOKE REACTOR

SLOWPOKE is the name of a 5 kW pool-type nuclear reactor recently developed at Chalk River. After being in operation for a year it has now been transported and placed in operation at the University of Toronto.

Figure 1 is a view of the reactor container prior to being lowered into a water filled pool to provide shielding. The University of Toronto will use Slowpoke as a source of neutrons to make short lived isotopes and for neutron activation analysis.

The reactor exploits a recent discovery made at the Los Alamos Scientific Laboratories: that a surprisingly low mass of enriched uranium can be made critical using a beryllium metal reflector. As the ratio of thermal neutron flux to fission power is approximately inversely proportional to critical mass of $^{235}$U in the reactor, it is possible to obtain a high flux with relatively low fission power. The cover of this issue shows the core full scale, dramatically illustrating its small size.

A significant feature of the reactor is its large negative temperature coefficient of reactivity. This means that reactivity decreases with increasing reactor temperature. Most reactors have this characteristic; but in Slowpoke it is exploited so that all conceivable perturbations from reactivity changes result in safe, slow power changes. This feature has led to the elimination of conventional instrumented protective systems. Instead reliance is placed on this inherent, and hence highly reliable physical phenomenon.

Slowpoke is probably the first nuclear reactor designed to operate unattended without telemetry.* Only periodic surveillance is made by a part-time operator to ensure that the few essential systems are operating correctly.

The challenge of reactor engineering is that the problems are interdisciplinary in nature. Slowpoke was no exception, and in addition to reactor physics, mechanical and electrical engineering, it involved mathematical, metallurgical and chemical problems, some of which are discussed below.

Heat Transfer

Light water flows by thermosyphon action through the core removing heat from the fuel by natural convection. Due to uncertainties in calculating natural convection steady state conditions, a full scale replica was built and tested. Orifices were sized to achieve desired core temperatures at the designed thermal power. These tests required exacting measurements, as temperature differences in natural convection heat transfer are small.

To ensure safe operation at high power the critical heat flux (CHF) of the fuel elements was determined. When the CHF is reached the dryout phenomenon occurs**, leading directly to fuel melting. Various correlations based on experimental results for low pressure pool boiling conditions were examined and a correlation deduced. Peak heat flux from the most severe power transient was shown to be less than 10% of the prediction CHF.

Reactor Dynamics

Computer simulations of the dynamic response of the reactor to reactivity changes were of great interest and formed the basis for predicting the degree of stability and safety prior to construction. Simulations

* At the University of Toronto, one radiation monitor has a remote alarm in another building.

**Dryout was a topic discussed in the Winter 1971 issue of "AECL Research in Engineering".
proceeded through three stages. At first a simple mathematical model was made using reactor kinetic equations with six groups of delayed neutrons and a simple differential equation to describe heat transfer from the core. These equations were solved by the GENASYS computer code, a program for analyzing system stability, and was run on the Bendix G-20 computer at Chalk River.

Results showed the system to be stable from measurements of the open loop phase margin and the decay time of power oscillations. However, the regulation, as measured by the power overshoot and decay time from a reactivity step input was found to be unsatisfactory. Therefore a second, more elaborate model was constructed which separated the fuel and moderator and took into account the non-linear effect of increasing reactivity feedback with temperature rise. Encouraging results were obtained which indicated the reactor would respond slowly and safely to large sudden increases in reactivity. Prior to the reactor going critical, a third, more elaborate model was made for sub-cooled operation. The reactor kinetic equations remained unchanged, while the thermal effects were divided into core, beryllium and water reflector sections. They were solved using the MIMIC code on a CDC-6600 computer. Figure 2 shows the close comparison between a simulated transient and a test carried out on the reactor. Further tests were made to 30 times rated power with no deleterious effect to the core, thereby demonstrating that the electro-mechanical protective system was not required. Elimination of this system was the single most important factor in simplifying the design and operation of the reactor.

Materials and Chemistry

The reactor fuel is made from extruded uranium-aluminum alloy clad with aluminum for improved corrosion resistance. Similar fuel has been successfully used for many years in the research reactors at CRNL. A modest development program was undertaken to weld aluminum studs directly to the ends of the elements using a capacitor discharge welding process. The objective was to obtain a sound peripheral bond between the cladding and stud. As the cladding is only 0.020 in. thick, the development proved difficult. Figure 3 is a metallographic sample of a typical stud weld. Figure 4 is a 100 times enlargement of the stud to cladding bond showing
unacceptable defects at the weld interface. Many test welds were made before an acceptable quality level was obtained in the production run.

Materials compatibility, corrosion and water chemistry were carefully considered during the design. The reactor was largely constructed from aluminum alloys, the reflector of beryllium, with stainless steel fasteners and small components. Water conductivity, pH and impurities were controlled by purification and chemical addition. When the reactor was dismantled in Spring 1971 after a year's operation, no evidence of deleterious corrosion was found.

Even a modest innovation demands new basic information. Such information inevitably cuts across boundaries of technology and demands simultaneously both broadness and depths of those seeking it.

P.D. Stevens-Guille

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HEAVY WATER PRODUCTION USING AMINE-HYDROGEN ISOTOPIC EXCHANGE

Development of the amine-hydrogen exchange process for heavy water production involves a large research and development program at CRNL supplemented by contracts with industry and universities. Four major areas of work are: process design and optimization, process chemistry, deuterium exchange in gas-liquid contractors, materials of construction.

This article, which is the second in a series to appear in this publication, describes some aspects of the chemistry of the process. The first article (1) gave some of the background for using and producing heavy water and described the process flowsheet.

The amine-hydrogen process being developed at CRNL is a two-temperature process similar to the $\text{H}_2\text{S}-\text{H}_2\text{O}$ process used for three large plants in Canada, but based on the equilibrium reaction

$$\text{CH}_3\text{NH}_2 + \text{HD} \rightleftharpoons \text{CH}_3\text{NHD} + \text{H}_2$$

The reaction is catalysed by potassium methylamide ($\text{CH}_3\text{NHK}$) dissolved in the aminomethane. Because the equilibrium constant decreases with increasing temperature, it is possible to have deuterium transfer from hydrogen to a countercurrent flow of liquid aminomethane in a cold tower and reverse this transfer in a hot tower. The deuterium concentration reaches a maximum at a location between the towers. A small stream is taken from the point to a successive enrichment stage. (For more details see the flowsheet in the previous article.)

An advantage of the amine process over the water-hydrogen sulfide system is the possibility of using lower temperatures because of the low freezing point of the aminomethane, $-93^\circ\text{C}$ compared to $+30^\circ\text{C}$ for the $\text{H}_2\text{S}$ hydrate.

The greater difference in equilibrium constants between hot and cold regions (see table below) allows much more effective separation of deuterium and hydrogen, and correspondingly lower flows.

| EQUILIBRIUM CONSTANTS FOR $\text{H}_2\text{S}-\text{H}_2\text{O}$ AND AMINOMETHANE-$\text{H}_2$ SYSTEMS |
|---------------------------------|-----------------|-----------------|
| Cold Region                     | Hot Region       |
| $-40^\circ\text{C}$             | $-70^\circ\text{C}$ | $-70^\circ\text{C}$ | $+130^\circ\text{C}$ |
| $+30^\circ\text{C}$             | $+130^\circ\text{C}$ | $+130^\circ\text{C}$ | $+130^\circ\text{C}$ |
| Amine-$\text{H}_2$              | $8.6$            | $4.2$            |
| $\text{H}_2\text{S}-\text{H}_2\text{O}$ | $2.2$            | $1.8$            |

Process chemistry includes kinetics of the exchange reaction, preparation of the potassium methylamide catalyst, catalyst solution stability and establishment of analytical techniques applicable to the system. It is necessary to recognize that the solution, potassium methylamide in aminomethane is non-aqueous, highly reactive and a potentially hazardous material, requiring special handling techniques.

The kinetics of the exchange reaction are important because they determine how quickly equilibrium between the amine and hydrogen is attained and therefore affect the size of the exchange towers. It is desirable to have the reaction rate as fast as possible. Aminomethane is preferred over ammonia because of its faster reaction rate.
A laboratory scale apparatus is used to measure the rate under various conditions and to evaluate the effect of potential additives or of impurities in the solution. A schematic diagram of the exchange cell is shown in Figure 1. Solution to be tested is put into the cell along with hydrogen gas previously enriched with deuterium. A magnetically driven stirrer at 3000 rpm provides intimate mixing of the hydrogen with the liquid phase, where the exchange takes place. A direct connection between the exchange cell and a mass spectrometer provides deuterium analysis of the gas as a function of time. The results from an exchange experiment are shown in Figure 2. The exchange rate constant is calculated from the decrease in the deuterium concentration of the gas phase; in the case illustrated the rate constant is 45 min⁻¹.

The catalyst is formed by reaction of potassium metal with aminomethane

\[ K + CH_3NH_2 \rightarrow CH_3NHK + 1/2H_2 \]

In principle this procedure is simple, however when the two reactants are brought together they react very slowly. Since large batches of catalysts are required in a plant, a test facility was built to determine the best way to make the catalyst. This rig (see Figure 3) provides means of circulating the amine at different temperatures and for adding potassium metal as either a solid or liquid. The catalyst formation involves at least two steps; dissolution of the potassium metal, followed by reaction with the sol-

vent to produce potassium methylamide and hydrogen. We found that dissolution was the slow step and that good agitation and high surface area of the potassium were necessary to prepare the catalyst in a reasonable time.

Stability of the catalyst solution refers to its reactions with impurities that may contaminate the system, as well as its thermal stability. Potential feed gas impurities are air, H₂O, CO, and CO₂; all react with and destroy the catalyst, reducing the exchange activity of the solution. The reactions are:

\[
\begin{align*}
CH_3NHK + H_2O & \rightarrow CH_3NH_2 + KOH \\
CH_3NHK + CO & \rightarrow HC - NK (N\text{-methylformamide}) \\
CH_3NHK + CO_2 & \rightarrow CH_3NHCO\Theta + CH_3NH_3 \\{\text{Carbamate Salts}\}
\end{align*}
\]

With oxygen the reaction is complex and all the products are not identified. Potassium cyanide is one product.

Potassium methylamide-aminomethane solutions thermally decompose, with reduction in exchange efficiency. The decomposition is rapid at 100°C but the solutions are stable at lower temperatures and the stability is increased by a high partial pressure of hydrogen. Thermal decomposition is not a serious
process restriction. The initial step of thermolysis is believed to be dehydrogenation of potassium methylamide

\[ \text{CH}_3\text{NH}_2 \xrightarrow{\Delta} \text{CH}_2 = \text{NK} + \text{H}_2. \]

In parallel with the process chemistry studies there continues to be extensive analytical development. A method of automated, on-line mass spectroscopy analysis for deuterium in hydrogen is established, as well as procedures for deuterium in the amino group of aminomethane. Analysis of these non-aqueous, reactive solutions has required development of many conventional methods which are normally applied in aqueous systems.

These studies have revealed a chemical system that is largely unexplored and could be a fertile field for fundamental chemical research. The mechanisms of many of these reactions and the analytical techniques for non-aqueous systems need detailed investigation. While the application of the hydrogen-amine process is not dependent on immediate answers to these questions, the work must be done to ensure the understanding necessary for optimum process performance. Our program will also begin studies of process chemistry behaviour in pilot systems where conditions better simulate a production plant.

W.J. Holtslander

REFERENCES


Additional information can be found in the following documents:
H.K. Rae, “Heavy Water in Canada”, AECL-3866 (1971)
A NEW TECHNIQUE FOR FREQUENCY RESPONSE MEASUREMENTS

The design of a control system for a large nuclear power station requires a thorough knowledge of the dynamic characteristics of all components in the system and how these interact.

To predict system behaviour the designer must have mathematical models for the response of each component or subsystem. These models may be developed analytically or from manufacturing specifications. In either case it is often necessary to verify the validity of the mathematical model for certain critical subsystems. Furthermore, some components may require testing after they have been in service, to determine whether there has been deterioration in performance.

One of the most useful measurements of a system's dynamic performance is its frequency response. A new method of measuring frequency response has recently been explored at CRNL and a practical technique developed [1]. In this method a component is perturbed with a filtered pseudorandom binary sequence and a fast Fourier transform is used to determine the frequency response from the measured time response. Figure 1 is a block diagram showing how a measurement and analysis is performed. Records of typical input and output signals from a simulated component are shown in Figure 2. Before Fourier transforms of these signals can be obtained by the computer they must be converted to

Figure 1 Measurement of frequency response using a pseudorandom binary input signal.

Figure 2 Input and output of simulated component.
digital form. As each conversion is done the result is stored in a digital memory until the required number of points has been collected. Figure 3 shows a complete block of input and output data. The Fourier transform is taken to convert input and output data blocks from a sampled time series to a discrete Fourier series. The ratio of output to input for each frequency is now calculated. The familiar form of representing frequency response, magnitude and phase angle versus frequency, is then computed and yields an output as shown in Figure 4.

The technique can be used to check the performance not only of individual components but of whole systems and should be able to predict the onset of undesirable operating characteristics long before human observation could detect it. Furthermore, tests can be performed without interfering with the normal operation of the system, a most desirable feature when plant availability is important.

C.B. Lawrence

REFERENCE


*
UP-RATING THE U-1 LOOP

Experimental nuclear fuel is best tested in "loops" in which coolant conditions can be controlled independently of those in the rest of the reactor. Flow, temperature, pressure, coolant chemistry are all independently variable and the more rearrangeable each of these is, the more useful the loop is apt to be.

Among AECL's experimental reactors NRU offers the biggest lattice sites and the highest neutron fluxes and can best accommodate loops for irradiating full scale fuel bundles. Of the three big loops in NRU, U-3 uses an organic coolant, U-2, pressurised water, and U-1, two phase water.

The loop is intended to operate in one of four modes—pressurised water, boiling water, fog and steam (the distinction being that in the first mode the inlet and the outlet would both be sub-cooled; in the second mode only the inlet would be sub-cooled; in the third mode both inlet and outlet would see wet steam and the fourth the outlet would be superheated). Figure 1 shows a greatly simplified flow sheet illustrating piping changes that had to be made in various modes. It will be seen that pressure control was always obtained using steam from the boilers. The fog and steam modes both cool the fuel with

---

Figure 1 Simplified flow diagram of U-1 loop.
steam, but the vastly better heat transfer to wet steam means that in the fog mode, much more powerful bundles can be irradiated, requiring much more boiler steam. When U-1 was originally designed in 1964, the fuel bundles of interest were only 3 1/4" dia. In those days a boiler of 1.6 MW capacity was considered adequate. The maximum heat rejection capacity was 4 MW. Before the loop was commissioned the decision was taken that future PHWR and BLW CANDU reactors would use 4" dia. bundles. The addition of nearly 50% more fuel required the addition of nearly 50% more steam, if a design range-ability was to be maintained.

The boilers selected for the loop are electrically fired natural convection devices with a central down-comer and hairpins of mineral-insulated metal-sheathed heaters in an annular riser. There were originally eight such boilers with 8" shells in parallel with one separator and a single level control. The control of level is obviously critical since the boilers are very short. (It is also important to minimise the possibility of the heater sheath defecting since at plausible heat fluxes, contact between the magnesia insulant and water results in violent expulsion of the magnesia which deposits on the nuclear fuel).

After the loop had been in service for some few months a heater defected and an examination of the sheath suggested stress corrosion cracking as a mode of failure. Since there was evidence that leachable chloride in the boiler crud may have caused similar cracks in all the heaters, it was decided to replace them all and at the same time to increase their resistance and put the power up to 2 MW. This entailed relatively little change except the addition of a new power supply on the roof. Some part of the total heater power is modulated by a saturable core reactor. The rest can be switched in blocks, and it was not felt that more continuous modulation was needed.

Experimental programs in any loop must necessarily expose the fuel to a larger risk of failure and radiation than exists in a power reactor. In fact tests of deliberately defected fuel are a necessary part of the program, since the behaviour of escaping fission products must be known before failed fuel monitoring systems can be designed. This means that on occasion quite high radiation fields are developed in various components. Loop rooms are therefore quite heavily shielded. Figure 2 indicates the structure of the loop rooms in NRU. The clear plastic in this 1/8" scale model represent something like 4 ft of concrete in the walls and ceilings. These rooms were something of an afterthought in the design of the building, and although they are generous in size by comparison with those in NRX, the complexity and the size of the circuits in NRU make the rooms extremely crowded. Figure 3 shows a typical part of the full scale installation. Finding room for more boilers and coolers is obviously a problem.

Since NRU is only 10 ft long and power reactors at least 16 ft it is not possible to test a full length fuel channel. The conditions at any one critical point in a channel, for instance the point of closest approach to dry-out, can be modelled by reducing the flow and thus increasing the local quality. With an increased interest in heat transfer close to and beyond dry-out, it has become important that we should be able to
achieve mass velocities, comparable to those in a power reactor channel, which requires another 2 MW of steam.

The major problems in doubling the boiler power have been space and floor loading. The structural integrity of the concrete loop rooms is already fairly strained and indeed, there has been some cracking although this has been attributed primarily to thermal stress (the loop room runs sometimes at 120° ambient). The space and weight limitations were solved jointly. A lead wall alcove, containing instruments which need daily maintenance was removed, and the instruments were located outside the room. The thick concrete roof hatch was also removed and replaced by a lead covered steel hatch from which was suspended a high velocity room cooler, nesting in the hole left by the concrete hatch.

Figure 3 Typical detail of field installation.

<table>
<thead>
<tr>
<th>Mode</th>
<th>Press. Water</th>
<th>Boiling</th>
<th>Fog</th>
<th>Steam</th>
</tr>
</thead>
<tbody>
<tr>
<td>Flow (kg/s)</td>
<td>12</td>
<td>12</td>
<td>12</td>
<td>3</td>
</tr>
<tr>
<td>Press (bars)</td>
<td>100</td>
<td>100</td>
<td>100</td>
<td>100</td>
</tr>
<tr>
<td>Inlet Temp (°C)</td>
<td>280</td>
<td>280</td>
<td>*</td>
<td>450</td>
</tr>
<tr>
<td>Fuel ΔP (bars)</td>
<td>9</td>
<td>9</td>
<td>9</td>
<td>9</td>
</tr>
<tr>
<td>Outlet Temp (°C)</td>
<td>*</td>
<td>*</td>
<td>*</td>
<td>600°C</td>
</tr>
<tr>
<td>Fuel Power (MW)</td>
<td>4</td>
<td>5.5</td>
<td>5.5</td>
<td>1.5</td>
</tr>
</tbody>
</table>

* Saturation Temperature at Local Pressure

Figure 4 shows a model of the revised layout. A second set of eight boilers has been added in a package geometrically similar to the existing one. A new helical coil cooler and a third pump have been added to the heat rejection system. This leaves some space near the entrance to the room. If in the future it is decided to add a superheater to the boiler outlet it will be located here. Until that time comes, we propose to install a steam generator in parallel with the 4 MW electric boiler using hot organic liquid from U-3 to make steam from the U-1 boiler feedwater. This steam generator, if it is approved will be strictly an alternative source of power to the electrical boiler system. The loop operators will be able to select one or the other. Eventually we hope to propose a second system which will reject U-1 loop heat to the plant steam system.

When these modifications are made the capability of the loop in each of its modes will be as shown in Table 1. It will be one of the largest in North America. As well as testing the nuclear performance of prototype and development fuel bundles it offers a unique site for very high heat flux thermodynamic experiments.

B. Godden

*
GROWTH OF RELIABILITY THROUGH DEVELOPMENT

One object of development is to improve reliability of mechanical components. The purpose of this note is to show how quantitative estimates may be made of the increase of reliability during a development program. For this purpose a probabilistic model is developed which relegates subjective judgments to inputs.

Our model, first considered by Lloyd and Lepow\(^1\), has a component with two modes of failure which we shall call "inherent" and "assignable cause". Each test has the following possible outcomes, success, inherent, or assignable cause failure. At each assignable cause failure one attempts to repair the component and remove the cause of failure. The probability of removing the cause of failure may differ from trial to trial and we can include a learning process which increases the probability of successful repair with increasing trials.

We assume the following:

- Probability of an inherent failure = \(q\).
- Given component in unrepaired state and an inherent failure does not occur, probability of assignable cause failure = \(q\).
- If component is in repaired state probability of assignable cause failure = 0.

\(E_{0,n-1}\) = component is unrepaired after \(n-1\) trials
\(E_{1,n-1}\) = component is repaired after \(n-1\) trials.

Now we can express the reliability of the component after the \(n\)th trial as:

\[
R_n = \text{Probability of no inherent and no assignable cause failure, given the component is in state } E_{0,n} \text{ times probability system is in state } E_{0,n}^* + \text{Probability of no inherent failure given system is in state } E_{1,n} \text{ times probability system is in state } E_{1,n}^*
\]

or

\[
R_n = (1 - q)(1 - q)^n \cdot (E_{0,n}) + (1 - q)(1 - p(E_{0,n}))
\]

Examination of (1) reveals that the maximum reliability is \(R_n = (1 - q)\) which, as to be expected, is determined by the inherent failure rate, and that development should concentrate on reducing the probability of the system being in State \(E_{0,n}^*\). Hence we will consider how development can effect \(P(E_{0,n})\).

A natural development procedure is to attempt a repair at every assignable cause failure. The probability of effecting a repair at the \(u\)th failure is \(b_u\). We expect \(b_u\) to increase as our knowledge and experience increase with the number of tests. Not every test will produce a failure so we may perform \('n'\) tests and observe \('u'\) assignable failures.
Now, the probability of being in State $E_0$, at the $n^{th}$ trial is

\[ P(E_0, n) = P(E_0, n-1)[P(u)(1-b_u) + q_o + (1-q_o) (1-q)] \]

Now 'u' may have all values from 1 to $n$, hence (2) is properly written as a sum, i.e.

\[ P(E_0, n) = P(E_0, n-1) \sum_{u=1}^{n} P(u)(1-b_u) + q_o + (1-q_o) (1-q) \]

(3)

To obtain an expression for $P(u)$ we observe that the probability of the $u^{th}$ failure at the $n^{th}$ trial is the union of an assignable failure at the $n^{th}$ trial and $u-1$ assignable cause failures in $n-1$ trials.

Given that the component is in the State $E_0$, the probability of an assignable cause failure is $q_o(1 - q_o)$, which we will denote by 'g'. Simple combinatorial analysis will show that

\[ P(u) = g \binom{n-1}{u-1} g^{u-1} (1-g)^{n-u} \]

(4)

Equation (3) may now be written as

\[ P(E_0, n) = P(E_0, n-1) \sum_{u=1}^{n} \left[ g^{u-1} (1-g)^{n-u} \right] \left[ (1-b_u) + 1 - g \right] \]

(5)

Equation (5) relates the probability of the component being in State $E_0$, at the $n^{th}$ trial if the system was in State $E_0$, at the $(n-1)^{th}$ trial. One sees readily that by the recurrence relation (5) $P(E_0, n)$ can be related to the initial State $P(E_0, 1)$ by the continued product.

\[ P(E_0, n) = P(E_0, 1) \prod_{j=1}^{n} \sum_{u=1}^{j} \left[ g^{u-1} (1-g)^{j-u} \right] \left[ (1-b_u) + 1 - g \right] \]

(6)
Equation (6) describes the change in the probability of the component being in the unrepaired state (assignable cause failure not corrected) as the development program proceeds. The objective of the program is to reduce this probability as rapidly as possible.

The subjective inputs are:

- $P(E_{o,0})$ = initial probability of being in unrepaired state,
- $q$ = probability of assignable cause failure,
- $q_o$ = probability of inherent failure,
- $b_u$ = probability of a successful repair.

Numerical results are given in Table I for the following processes.

<table>
<thead>
<tr>
<th>Process Without Learning</th>
<th>Process With Learning</th>
</tr>
</thead>
<tbody>
<tr>
<td>$P(E_{o,0}) = 0.8$</td>
<td>$P(E_{o,0}) = 0.8$</td>
</tr>
<tr>
<td>$q = 0.8$</td>
<td>$q = 0.8$</td>
</tr>
<tr>
<td>$q_o = 0.1$</td>
<td>$q_o = 0.1$</td>
</tr>
<tr>
<td>$b_u = 0.1$</td>
<td>$b_u = 1 - e^{-(0.5n - 3946)}$</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Trial Number</th>
<th>Probability of Being in Unrepaired State</th>
<th>Reliability</th>
</tr>
</thead>
<tbody>
<tr>
<td>n</td>
<td>No Learning</td>
<td>Learning</td>
</tr>
<tr>
<td>1</td>
<td>0.742</td>
<td>0.742</td>
</tr>
<tr>
<td>2</td>
<td>0.689</td>
<td>0.499</td>
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<td>3</td>
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</tr>
<tr>
<td>5</td>
<td>0.550</td>
<td>0.042</td>
</tr>
</tbody>
</table>

Figure 1 shows the increase of reliability for the two assumed programs of development and indicates how identification of causes of failure can mean the difference between success and failure of a program. Considerations such as these lead AECL to offer its engineers and scientists instruction in the Design of Experiments which stressed the isolation of effects.

W.A. Wolfe

REFERENCE:

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