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June 14, 1968

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DOUGLAS UNITED NUCLEAR, INC.

BY

STAFF OF DOUGLAS UNITED NUCLEAR, INC.

June 14, 1968

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DOUGLAS UNITED NUCLEAR, INC.

INTRODUCTION

The annual report on the previous fiscal year's accomplishments attributed to process development has been provided about six months after the end of the fiscal year. In accordance with a recent request, the date of this report has been advanced to coincide with the end of the fiscal year. Further, the scope of the report has been expanded to include a narrative assessment of program accomplishments. Because of this enlargement of scope some liberty has been taken in identifying new products and development gains.

The Douglas United Nuclear process development program can be categorized into the following: 1) Cost Reduction, 2) Increased Plant Life, 3) Versatility, and 4) Safety and Licensability. The overall program goal is to develop technology that will provide greater efficiency at reduced cost and yet be responsive to an increasing variety of product requirements while continuing the excellent plant safety performance. The goal is significant when considered in view of increasing budgetary pressure. The projected defense requirements after 1975 are not expected to require the full capacity of the Hanford reactors. However, based upon the anticipated growth of electric power requirements and space exploration power requirements, the current full productive capacity of the Hanford reactors will not meet the projected breeder reactor fuel and/or isotope requirements. Thus, process development in support of current and near-term defense requirements will find application in the longer-term nondefense market.

It will be apparent that, in many areas, a direct dollar savings is not applicable inasmuch as the demonstration of versatility and/or increased plant life is valuable only in a relative sense.

COST REDUCTION

1. N Reactor Fuel Performance

Fuel design and fabrication parameters, which were identified as limiting the performance of Mark I fuel to slightly more than design conditions of 4000 MW reactor power level and 2100 MWD/T goal exposure, have been modified in the design of the Mark IV fuel element. The Mark IV fuel, a nominal 24 lb/ft, will incorporate end closure and support modifications along with the use of an iron-aluminum-uranium alloy. These modifications are expected to virtually eliminate the known causes of Mark I fuel failures. Also, redimensioning the inner and outer fuel pieces will provide a more balanced coolant enthalpy thereby permitting higher tube powers. In addition to the specific cost savings associated with these improvements, the successful development of the Mark IV fuel is expected to provide for routine operation at reactor power levels up to 4800 MW.

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2. N Reactor Power Level Stretch Capability

The successful probe of power level stretch capability conducted during the co-product program demonstrated the inherent capacity of the primary and secondary systems which permitted the reactor to be operated with six cells at 4800 MW. During the initial phases of preparing for the reactor power level probe, it became apparent that, if some degree of primary coolant boiling could be permitted, a sizable advance in thermal and electrical power level could be made. Such an advance could exploit the N Reactor's steam revenue capability. Further system analysis and engineering study showed that reactor operation with coolant boiling in a portion of the fuel channel was feasible and that reactor power levels of 5600 MW could be realized with only modest equipment capital investment. The electrical potential was determined to be between 1150 and 1200 MW, which would require a third generator at the WPPSS plant. The technological development work required for this advancement has been identified and an effort is in progress to provide confirming data. Further, specific plant and equipment modifications to accomplish power levels above 5200 MW have been identified.

3. Computerization

Modifications to the ISI-609 data logger at N Reactor have been accomplished which effectively converts the equipment to data processing. The utility of the equipment was amply demonstrated and forms the basis for developing an on-line process computer and ultimately closed circuit automated control. The benefits to be derived from such developments are reduction in manpower requirements for prestartup checkout, process limit monitoring, margin computation, and improved assurance of compliance with nuclear safety requirements.

Preliminary scoping and engineering studies were carried out to define the incentives and requirements to initiate computerized control at the K reactors. It would appear that the incentives are high and the program should proceed with additional emphasis. This is particularly true with respect to the improved nuclear safety aspects and would provide even greater assurance in the event of higher power density fuel loadings.

4. General

A key requirement for design of Mark IV fuel was an increase in fuel weight by approximately 14 percent, with the result that the dimensions of coolant flow annuli were reduced to significantly less than those of Mark I fuel supports. This requirement, together with a 20 percent increase in design power level, resulted in the need for an outer cylinder support system that would cause considerably less pressure drop than Mark I supports. A simple and economical solution was developed through theoretical analysis and hydraulic testing. The results showed that the required pressure drop and structural characteristics could be obtained by rearranging the location of the already developed Mark I suitcase handle supports. This eliminated the need for a comprehensive support redesign effort that was judged to have questionable prospects for success. Lack of success in producing a satisfactory new support design would have required

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extensive out-of-core hydraulic tests, charge-discharge tests, and in-core production tests, resulting in a six to nine-month delay in the Mark IV program or redesign of the fuel which would have resulted in the same or greater delay and would have required additional engineering and manufacturing efforts.

Savings: \$251,000

Limits were developed to permit larger quantities of fuel to be loaded in the NPR cask and thereby lowered the number of yearly shipments. Cost savings were realized for the 2.1 w/o U-235 coproduct demonstration load and a tube-in-tube reactor load.

Savings: \$ 58,000

Limits were established that allowed more flexible charge-discharge operation. The need for frequent setup and discharge cycles was eliminated. Most significant was the lower time required to charge the spike component of the reactor load. The modified procedure and revised criticality limits result in an annual saving of 22 outage hours; no dollar savings are shown since it is difficult to assess, except in relative terms.

The 184 N boiler performance was tested at low steam flows and, as a result, the allowable minimum load was reduced from 30,000 to about 10,000 lb/hr.

Annualized Savings: \$120,000

Fuel spacers were found to cause damage to process tubes and were modified.

Savings: \$ 22,000

Special procedures were developed to permit movement of steam generator tubing into the reactor heat dissipation cells without requiring construction of a special air lock.

Savings: \$ 40,000

In-reactor tests showed that Mark I fuel could be operated at higher tube powers by:

1. Better centering the fuel in the process tubes,
2. A slight redistribution of coolant among the cooling passages.

The changes were accomplished simultaneously by replacing the steel shoes on all Mark I fuel with thicker shoes (for centering) having small tabs or "wings" to increase pressure drop in the outer annulus. The refurbished fuel, designed Mark I-C, is now being irradiated successfully in the reactor. Essentially one and one-quarter loads of fuel were improved this way.

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All new production Mark I-C fuel is being made with double-pass welds which is intended to improve the integrity of the closures and avoid end-associated failures. The above modifications result in avoiding an incremental product loss as well as eliminated extra costs associated with three to five days outage time. The value of such incremental production depends upon the assumed value of plutonium, therefore, no quantitative dollar value is shown.

End-associated causes accounted for seven failures of Mark I inner fuel elements. A number of improvements have been made in Mark IV fuel that are expected to eliminate such failures. Chevron end caps are being used on all Mark IV production, both inners and outers. This type of end cap performed perfectly (no failures) on Mark II fuel (outer only). The Mark IV design has been balanced to provide equal uranium temperatures and heat fluxes in the inner and outer elements, thus lowering the temperature of the inner. A double-pass welding process is expected to further improve the integrity of the closure. If the improvements are successful (no failures have occurred in current testing which has reached 50 percent of goal exposure), then about four failures per reactor loading will be avoided.

Annualized Savings: \$150,000

Diligent analysis of Mark II fuel performance led to the discovery that localized overheating was causing failures. Impaired heat transfer at one point on the fuel, under the locked solid support, created excessive local temperatures on maximum power fuel elements. Discontinuation of the coproduct demonstration was averted by lowering local power and increasing subcooling. Another result of the analysis indicated the possibility of changing support contour so that a similar failure problem should now be avoided in that fuel.

Savings: \$480,000

A continuing program to investigate the corrosion behavior of the components of the C-K Reactors led to a reduction of the amount of sodium dichromate added to the reactor coolant as a corrosion inhibitor. The dichromate concentration at one-half of B Reactor, C Reactor, and KE Reactor was reduced during FY-1968 from 1.0 ppm to 0.5 ppm.

Savings: \$ 35,000

A half-plant test carried out in FY-1967 demonstrated that effective removal of the parent material which is irradiated to form the radioactivity associated with the C-K Reactor effluent is obtained when the flocculant (alum) feed rate is adjusted so that the electrokinetic charge on the floc particles is zero. Although at some times of the year this type of control results in alum feed rates above the 18 ppm used as a feed rate for several years, the effect of the change to control based on need at the plants resulted in a savings of \$32,000 in FY-1968.

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INCREASED PLANT LIFE

1. Graphite Studies

The irradiation of graphite removed from a K reactor continued in the high flux zone of the GETR. At temperatures above those normally experienced in the K reactors, the point at which contraction stops and expansion begins was identified. The irradiation will be transferred to the ETR to define the point of turnaround at temperatures more nearly representative of K reactor operating conditions as well as the rate of expansion. The effects of the graphite expansion on the K reactor stack are also being studied.

2. Zircaloy Process Tube Hydriding

The mechanism causing the formation of the hydride case layer on the downstream section of the K reactor Zircaloy process tubes was defined, and corrective action was taken to prevent further buildup by the use of anodized spacers. It was also discovered during the year that there apparently is a second reaction taking place which transfers hydrogen from the coolant into the base metal even though the hydride case layer is no longer increasing in hydrogen concentration. Inasmuch as an increase in base metal hydrogen content will cause tube failure at some point in time, studies to define and resolve the second mechanism have been initiated and are expected to continue through FY-1969.

INCREASED VERSATILITY

1. High Pu-240 Content Plutonium

Both K reactors operated throughout the year to produce plutonium having a 25 percent Pu-240 content by irradiating depleted uranium. While this program can not be considered to be of a process development nature at this stage, it is the fruition of research and development efforts carried out in prior years.

At the request of the AEC, the production of plutonium containing 9-12 percent Pu-240 was carried out in B and C Reactors. During this campaign, which was again the result of earlier research and development work, tests were carried out to evaluate both the corrosion behavior of aluminum fuel cladding as well as the stability of uranium fuel cores. It was shown that uranium core swelling would limit future irradiations in the neighborhood of 2000 MWD/T.

2. Tritium Production

The completion of the coproduct demonstration program has provided on-the-shelf technology for the coproduction of tritium and weapons-grade plutonium. In accomplishing this program, numerous supporting technologies were developed which enhance the flexibility and economic position of the AEC in the special nuclear field. The technologies developed include:

1. Process and fabrication technology for ceramic targets including development of manufacturing source of LiAlO_2 ,

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2. Extraction process for recovery of tritium from ceramic targets,
3. Fabrication and handling (including critical mass technology) of modestly enriched fuel (2.1 percent U-235) throughout the fuel chain,
4. Pilot scale (approximately 10 columns) demonstration of feasibility of coproduction of nondefense plutonium and tritium.

These technologies are a direct result of the coproduct program; however, much of the success is attributable to the status of a number of basic development activities in the areas of fuel development, thermal hydraulics, and physics.

A small-scale irradiation test in which lithium targets were driven by 210 Metal was carried out at a K reactor. The success of the test and subsequent safety documentation have brought the single-pass reactor 210 E-N program to the point of large-scale demonstration. A quarter reactor test has been planned, and it is expected to be carried out in FY-1969.

3. Other Isotopes

A process was devised and demonstrated to irradiate Np-237 for the production of Pu-238 with an out-of-reactor time on the order of 45 to 50 days. Such a process would maximize the utilization of Np-237 and result in substantial cost savings in Pu-238 production. A proposal was made to the AEC recommending that the irradiation, separation, and recycle of Hanford produced neptunium be carried out at Hanford using the rapid turnaround process.

Preliminary irradiations were also carried out to investigate the use of graphite as the neptunium target matrix vice aluminum in order to minimize the Pu-236 content of the Pu-238. Investigations of graphite and other matrices will continue through FY-1969.

Process and economic analysis studies were performed on the optimum management of U-236, the precursor to Np-237. Utilizing government-owned tails of commercial reactor spent fuel to spike recycled fuel from Hanford, current fuel streams in N Reactor are now significantly higher in U-236 content. Approximately 110 tons of Mark I and 152 tons of Mark IV fuel are involved in the initial "spiked" material. The increase in Np-237 amounts to 36 percent and 39 percent, respectively. The overall effect of the U-236 management effort is to attain U-236 equilibrium concentration in the feed material to N Reactor about two years sooner than would be attained with normal recycle.

Small-scale irradiations of Co-59 and Tm-169 were carried out to demonstrate capability.

NUCLEAR SAFETY AND LICENSABILITY

The primary goal of the nuclear safety process development activities is to provide scientific and engineering data in support of a more realistic appraisal of status

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of nuclear safety at Hanford reactors. The results from these studies will provide added assurance that Hanford reactors are operated in compliance with current criteria and will assure compliance with more stringent regulations anticipated in the future. As such, then, new processes or new products do not directly result from this work; however, the development of new processes, new products, or new operational modes at the Hanford reactors to gain higher productivity or lower cost is basically dependent upon a thorough understanding of all aspects of the nuclear safety picture. These nuclear safety development programs are, therefore, aimed specifically at the unique features of the Hanford reactors and the Hanford site. Results from these programs have been and will continue to be directly applicable to the Hanford site and complementary of other R&D programs funded from other sources. Results from studies to date have been used to provide a basis for nuclear safety capital improvement programs (as in the soil waste disposal aspect of N Reactor's Effluent Control Program), to define allowable reactor power levels and instrumentation settings from measured fuel failure temperature and fission product releases, and to provide other basic experimental and analytical data essential for maximum utilization of the reactors at minimum cost. Studies of fission product release from Hanford-type fuels and the metal-water reaction studies using full size zirconium clad uranium fuel will yield results applicable throughout the AEC complex and nuclear industry as well.

Major accomplishments have been made in the metal-water reaction studies with testing now completed on full-sized unirradiated fuel and testing of irradiated fuels to start shortly. A program in meteorology has been established. Temporary instrumentation on the 300-foot tower is being used to obtain preliminary data at the N Reactor site, and data are being obtained on the Wahluke Slope to define meteorological conditions that affect the atmospheric movement of radionuclides. Both programs are expected to play a major role in establishing the technical basis for the equivalent of licensability at N Reactor. Studies to determine the efficiency and reliability of removing fission products by fixation on soil will provide basic data for hazards analyses which conclude that there is minimal potential for river contamination under reactor accident conditions.

Heat and radioactivity discharged to the Columbia River has come under intensive scrutiny from both State and Federal regulatory agencies. The concept of ground disposal of reactor effluent was studied by a test in B Area in which 50,000 gpm of effluent was discharged into an existing trench. Although the trench was near the river and had a relatively small surface to input flow ratio, some thermal degradation was apparent at the river bank and high decontamination factors were observed for Zn-65 and P-32.

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