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ANNUAL REPORT OF N REACTOR OPERATING EXPERIENCE
PERTINENT TO NUCLEAR SAFETY - CY-1973

AUTHOR

Staff of United Nuclear Industries, Inc.

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ANNUAL REPORT OF N REACTOR OPERATING EXPERIENCE
PERTINENT TO NUCLEAR SAFETY - CY-1973

February 15, 1974

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ANNUAL REPORT OF N REACTOR OPERATING EXPERIENCE
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ANNUAL REPORT OF N REACTOR OPERATING EXPERIENCE
PERTINENT TO NUCLEAR SAFETY - CY-1973

INTRODUCTION

This report is submitted to meet the requirements for annual reporting to the Commission of N Reactor operating experience pertinent to reactor safety.¹ Those CY-1973 accomplishments and experiences cited below have been selected as being responsive to the Reference 1 document.

PERFORMANCE CHARACTERISTICS

Rod System Operation

Except for isolated cases of component breakdown, the number of "slow" rods during a scram has been reduced to zero. This is the result of the supplementary spray lubrication of the exposed part of the actuating piston rods, which was initiated in November 1971. The supplementary lubrication makes up for the lower lubricity of the fire-resistant, water-in-oil emulsion that has replaced the turbine-type oil as the hydraulic fluid.

The neoprene wiper without the supplementary spray lubrication appeared to be the main cause of "slow" rods; however, there is some evidence that the elastomer section of the packing will bind if all the hydraulic fluid is sealed off from it. Those rods having the oil-soaked felt wipers are so few and so scattered that there is no incentive for eliminating the supplementary spray lubrications to the exposed actuating piston rods.

During the 1973 summer outage, nearly all the rod tip aluminum inserts containing the boron carbide nuclear poison were replaced. (See page 23.)

Ball System Operation

All ball system hoppers were tested at least once during CY-1973 to ensure maintenance of nuclear safety requirements for the system. During this period, two hoppers failed to trip on the first attempt, and a third hopper was found to have a slow trip response time. The two failures to trip are attributed to seizing of the ceramic balls on the exposed side of the hopper gate, and the

1. UNI-15, "Operating Safety Limits - N Reactor," Staff of United Nuclear Industries, Inc., June 1, 1973.

slow hopper response time was found to be a hopper solenoid failure, which was corrected by replacing the solenoid. Analysis of these failures indicates that no new failure modes have occurred and that the ball hopper trip experience meets the nuclear safety requirements. (See Unusual Occurrences pages 43 and 44).

Fuel Performance

The fuel failure history at N Reactor as a function of reactor production is shown in Figure 1 for the calendar years 1971, 1972, and 1973. Late in 1971, the need for reduction in fuel failures prompted the formation of a task force. This group developed short- and long-term programs to improve fuel performance. These programs were instituted during the period of early to mid-year 1972. Short-term results were apparent for the last half of 1972, when approximately 430,000 Mwd were produced with only four failures. Long-term results were even more significant in 1973, when 895,000 Mwd were produced with only three failures. The improved Mwd production per failure indicates a 490 percent increase for 1973 over 1971 and a 368 percent increase for 1973 over 1972.

The three failures that occurred in 1973 included two end-associated failures and one wire fretting failure. The wire fretting failure was typical of past wire fretting failures. The two end-associated failures, however, differed greatly. The first end-associated failure occurred after a very short five-day operating period. The failed element has been sent to the HEDL Radiometallurgy facility for examination. This examination is presently in progress.

The second end-associated failure was found to be an identified manufacturing defect that had not been removed from the manufacturing process. Unusual Occurrence Reports, reported on pages 54 and 55, cover the manufacturing and reactor aspects of this failure. As a result of this reject element failure, a records search was completed for all fuel in the reactor and in cold fuel inventory. A total of 20 elements in charged reactor fuel columns were identified as potential reject elements. These reject elements were discharged during the December 21, 1973, outage to remove the potential for additional failures from this cause.

CHANGES IN REACTOR OPERATION

Flow Monitor System Data Analysis

As reported earlier, computerized normalization techniques have been developed for comparing the indicated flow on each process tube with the average flow of other process tubes with similar flow characteristics, thereby providing a method for determining if the indicated flow of a given tube is in error. Work has continued toward refining these techniques to include the flow variations associated with special central zone enrichment columns, changing from "W" springs to "buggy" spring fuel element supports, changing the fuel element exposure in the reactor, and reactor flow distribution changes resulting

Figure 1. Fuel Failures as a Function of
Production for Calendar Years
1971, 1972, and 1973

from the various steam generator cell configurations. The data analyses have continued to be successful in analyzing and correcting flow indication problems associated with individual process tubes and in predicting flow monitor system limit settings following redistribution of reactor flow.

V-12 Diversion Capability

A report prepared by H. L. Debban, "Summary Report - V-12 Evaluation Group," dated March 14, 1973, described the status of divertability of the V-12 diversion valves in the rear pipe spaces of N Reactor. As a result of that study, V-12 leak repairs conducted during CY-1973 consisted largely of replacing leaking V-12 valves with spool pieces. A total of 92 spool pieces occupied V-12 valve positions as of December 31, 1973. UNI's plans for repairing V-12 valve leaks by replacement with spool pieces are described in References 1, 2 and 3. Removed V-12 valves are properly stored to permit future installation should full divertability be desired in the future.

One of the actions taken following the cladding failure of a fuel piece in process tube 2350 on December 7, 1973, was to review the requirements for diverting the flow of any suspect rupture fuel column from the primary loop. Later, in reply to a letter¹ from AEC-RL, one of the actions taken to limit concentration of I-131 entering the 1301-N crib was to eliminate the Process Standard requirement to divert flow of a suspect fuel cladding failure to the drain. This action was implemented by Process Change Authorization, "Rupture Detection and Control of Releases to the Crib," in the following sections:

- 1.c. "The diversion system monitors are no longer required to be in service."
- 2.b. "Process tubes for which there is a rupture indicated shall not be diverted to the diversion system. Operable V-12s on high exposure fuel and records of spool piece locations are no longer required."

Experience has shown that rupture debris does not result in large long-term increases in primary system component radiation levels. By not diverting the flow from process tubes with containing fuel failures, fission products within the primary system are not released to the environs until adequate means for their disposal are determined.

-
1. Letter, N. R. Miller to O. J. Elgert, "Use of Spool Pieces to Replace Leaking V-12 Valves," dated April 18, 1973.
 2. Letter, N. R. Miller to O. J. Elgert, same subject, dated November 15, 1973.
 3. Letter, N. R. Miller to O. J. Elgert, same subject, dated January 17, 1974.
 4. Letter, O. J. Elgert to C. D. Harrington, "Radionuclide Discharge Limits," dated December 27, 1974.

230 kv Power System Changes

In February 1973, the Bonneville Power Administration removed from service two 230 kv lines terminating on the Midway Substation bus. These lines were removed by BPA so that the lines could be rebuilt for 500 kv service. The two lines removed (Bonneville No. 2 and Grand Coulee No. 2) represented a generator-load source of power essential for Midway Bus Section No. 3 stability. Without these lines on Bus No. 3, there would be excess power swings of generators connected to Bus No. 3 following a Midway Substation Bus fault and opening of bus sectionalizing breakers. The modifications that were made by BPA to restore the power system stability of the Midway Substation Buses are as follows:

1. Moved Coulee No. 1, 230 kv line from Midway Bus No. 2 to Midway Bus No. 1.
2. Moved Bonneville No. 1, 230 kv line from Midway Bus No. 2 to Midway Bus No. 1.
3. Changed Midway protection relay tripping scheme from the tripping of all bus sectionalizing breakers for any bus fault to a system where only the bus sectionalizing breaker adjacent to the faulted bus would trip.
4. Installed a breaker failure tripping scheme which on failure of a bus section breaker to trip and clear a faulted bus initiated the tripping of all line feeder breakers and the tripping of the remote bus section breaker on the bus adjacent to the faulted bus.
5. Installed a tripping scheme that would initiate the tripping of all Priest Rapids lines for any Midway Bus three-phase fault. These changes restored transient stability to the Midway Bus Sections supplying the two lines to the Hanford 230 kv Loop and provided a bus fault clearing procedure with a backup fault clearing scheme.

DC Ground Troubleshooting

The result of UNI's continued effort to reduce the effect of 125 v dc grounds upon critical reactor control circuits was realized with no mandatory or accidental reactor shutdown during the year 1973.

A detailed study of each of the 145 critical circuits was completed to determine the effect of each circuit on plant operation if it were grounded. After each circuit's critical resistance was determined, its circuit breaker was categorized and color coded for easy, rapid identification.

Material was prepared and written, and troubleshooting procedures were established which were used for training 29 electricians and their supervisors in the classroom. A one-hour TV film was prepared for use by reactor operators for information, instruction, and training prior to their reactor certification tests.

New techniques, procedures, and test equipment were provided for detecting, isolating, and locating a ground on an energized circuit while the reactor continued in operation.

The Process Standards were expanded to include the additional information and describe the specific troubleshooting techniques that were permitted.

A new method of testing each circuit's integrity before startup after a prolonged outage was put into use to identify any circuit that has an impending problem. This was done with a 500-volt insulation tester to ground while the circuit remains energized.

The above mentioned methods, information, and equipment proved to be effective during 1973 in detecting, locating, and clearing six grounds during reactor operation, two of which required a reactor shutdown had they not been cleared. Four grounds were also quickly detected, located, and cleared during outages.

Recovery Actions for Fuel Handling Limit Violations

A review of nuclear safety procedures found there were no specific procedures for recovering from nuclear fuel handling limit violations.¹ As a result of the review, recovery action procedures were prepared and issued as PCA N-7-73. The procedures identify critical sections of Process Standard C-300, "Fissile Material Control," which involves one part of the double-contingency criterion. When violation of an identified critical limit is detected, all work in the area is halted and a broadly based Recovery Action Team is promptly convened. The team prepares a written plan of recovery, which is approved by the Directors of Reactor Operations and Technical Divisions and transmitted to AEC-RL Director of Production and Waste Management Division prior to implementation.

The critical sections of key manuals were identified prior to March 27, 1973. New pages with permanent markings for all Process Standard Manuals were circulating for approval at the end of the year.^{2,3,4,5}

The N Reactor Recovery Team was convened on one occasion associated with fuel inventory in the fuel basin. It was determined that a violation did not exist and that recovery action was not needed.

The Recovery Action Plan was placed in the Process Standards Manual on December 6, 1973.

-
1. Letter, T. A. Nemzek to L. M. Richards, et al, "Nuclear Safety Procedures," dated August 3, 1972.
 2. Letter, N. R. Miller to O. J. Elgert, same subject, dated September 15, 1973.
 3. Letter, N. R. Miller to O. J. Elgert, same subject, dated December 11, 1972.
 4. Letter, N. R. Miller to O. J. Elgert, "Criticality Safety Limit Violation Recovery Procedures," dated March 20, 1973.
 5. Letter, J. R. Bolliger to O. J. Elgert, "Nuclear Safety Procedures," dated June 25, 1973.

CHANGES IN FACILITY DESIGN

A number of changes in N Plant design that may be of interest are summarized in Appendix A.

UNUSUAL OCCURRENCES

Of the N Reactor related unusual occurrence reports submitted to AEC-RL during CY-1973, those considered to have nuclear safety connotations are summarized in Appendix B.

IN-SERVICE INSPECTIONSReactor Graphite Stack Surveillance

General - Graphite stack examinations are utilized to define required repairs to reactor ball channels and to assess the progressive effects of continued stack distortion in terms of anticipated damage to stack internals and future maintenance requirements. The following paragraphs summarize the related activities for CY-1973.

Ball Channels - The initial 1973 summer outage schedule included provisions for the examination of 13 ball channels. These examinations indicated that six of the inspected channels required "three-block repairs"¹ prior to being certified for subsequent service. Due to the high percentage of defective channels in the initial group, examinations were eventually extended to a total of 33 channels. Three additional defective channels were found, and the number of "three-block repairs" made totaled nine. The defective channels broadened the zone at the center of the reactor in which previous repairs have been made. Only one defective channel, No. 46, had previously required repair, that being a two-block repair which was conducted in 1971.

Channel 46 required special repair efforts due to accentuated upper channel curvature. The repair was successfully completed, however, and an inspection in December 1974 verified the integrity of the channel. A reaming tool is currently under development to permit some shaping of the channel which will both facilitate repair efforts under similar circumstances and correspondingly increase the expected period of adequacy for the repair. (See page 29.)

Channel 50 was locked out-of-service during the summer 1972 outage² due to concern over excessive clearances between the three-block repair replacement blocks and the bare channel walls. The circumstance arose due to the location of channel at the intersection of the front-to-rear and side-to-side graphite stack slip planes. Testing of the channel 50 configuration was conducted in the 189-D Tower ball channel mockup to determine if the unusually large clearances would affect the integrity of the replacement block junctions during ball drops. This testing proved that the current three-block repair technique was sufficient for this case. Therefore, channel 50 was scheduled in the original 13 channels inspected during the summer 1973 outage, and, after its integrity was verified, the channel was returned to service. (See page 29.)

-
1. "Three-block repair" consists of replacing the original top three liner blocks with ten modified replacement blocks better suited to accommodate the stack distortion induced channel curvatures which promote breakage of the original blocks.
 2. DUN-8112, "Annual Report of N Reactor Operating Experience Pertinent to Nuclear Safety - CY-1972," Staff of Douglas United Nuclear, February 15, 1973.

Twenty-nine of the 33 ball channels inspected during the summer 1973 outage were fitted with funnel liners. The funnel liner design and installations were authorized by Design Change 3249.¹ The funnel liner design was based on:

- Reestablishing the funnel-top liner block engagement which decreases as the liner blocks contract and settle away from the fixed funnel.
- Eliminating the need of extracting the original activated funnels to reestablish the engagement without sacrificing ball drop times.

Development testing of the funnel-liner design showed that the design decreased ball insertion times by about ten percent. Installation of the liners also afforded time and exposure savings compared to a complete replacement. This method will continue to be employed as required.

Rod Channels - Horizontal control rods 49, 59, and 69 were locked out of service in December 1972, subsequent to the failure of rod 59 to insert within the time specified by Process Standards. The insertion problem was due to the disengagement of a channel wall forming member or Tee Block, which became unkeyed due to vertical separations and relative block shifting which exists between graphite layers in the region of the top rod layer. Channels 49 and 69 were not blocked, but the potential for a similar occurrence in one or both of these channels was considered to be significant.²

The condition of the top row of the rod channels was reviewed during the summer 1973 outage based on additional rod channel examinations conducted at that time.³ The review concluded that channels 49, 59, and 69 should remain out of service. The potential for blockage in other top layer rod channels, however, was reverified as being negligible. Rod channels 48, 58, and 68 in the second layer of rods were also examined. The predicted absence of features essential to the channel blockage mechanism was confirmed.

Efforts are in progress to repair all top layer rod channels, except 59, during the summer 1974 outage. Development testing is in progress on equipment to enlarge the shield penetrations, broach the rod channels and insert graphite rod channel liner blocks. The effort is on schedule. The repair method will not apply to rod channel 59 due to its location on the side-to-side slip plane of the stack and the problems associated with keying blocks together in this location.⁴ No plans have been made to develop a repair technique for channel 59.

-
1. Design Change 3249, "Ball Channel Funnel Liner Installation," R. E. Russell, April 12, 1972.
 2. DUN-8112, "Annual Report of N Reactor Operating Experience Pertinent to Nuclear Safety - CY-1972," Staff of Douglas United Nuclear, February 15, 1973.
 3. UNI-51, "Recommended Operating Status of Horizontal Rod Channels," A. Russell, July 27, 1973.
 4. Letter, N. R. Miller to O. J. Elgert, "Program Plan - Control Rod Channel Renovation - N Reactor," dated January 14, 1974.

Graphite Cooling System Tubes - The proposed effort to develop GCS tube removal and replacement capabilities has been deferred due to the priorities assigned to rod channel renovation efforts.

The probing of 27 GCS tubes with plug gauges during the summer 1973 outage detected no significant progression of the tube distortion, confirming previously reported mock-up test data. The possibility of an isolated tube failure(s) due to localized block shifting is recognized. Preliminary analyses indicate that nuclear safety would not be impaired by isolated tube failures. The lack of repair capabilities to alleviate adjacent failures, however, could result in more restrictive specific power limits. Due to other priorities, this analytical study will not be completed until FY-1975.

Process Tube Elongation

N Reactor Zircaloy-2 30-percent-cold-work process tubes continue to elongate at the rate of 0.22 inches/year (4.5×10^9 kwhe). The elongation process believed to be irradiation-enhanced creep and/or anisotropic growth poses two problems:

- Only about one-half inch of axial expansion clearance, after accounting for thermal and pressure stress expansion requirements, remains before the tubes will begin to exert forces against the front and rear primary shields.
- The variation in rates observed between 30 percent and 17 percent cold-work tubes and due to integrated flux differences is altering clearances between the front face connectors.

Calculations indicate the front and rear reactor primary shields could accommodate approximately one-third of the total axial force from all tubes without failing. While several modification possibilities exist to provide additional expansion clearance, all are expensive in terms of manpower and personnel exposure. An alternative approach is, therefore, being investigated. The mechanism(s) of accelerated elongation observed on stressed Zircaloy-2 process tube material in a neutron environment has not been specifically defined as stress-dependent, irradiation-enhanced creep, stress-independent anisotropic growth, or a combination thereof. A test¹ has, therefore, been developed to measure the forces as a function of time by restraining a process tube and preventing its elongation, except for thermal and pressure expansion, and, instrumenting the assembly with a load cell.

If the elongation mechanism is strongly stress dependent, the reactor shields may have adequate strength to self arrest the process. Approximately two years are available to make this determination and, if necessary, to develop the capabilities necessary to provide additional expansion clearances. The test is scheduled to begin in March 1974, depending on reactor outage time availability and equipment development.

1. UNI-104, "PT-NR-349, Measurement of Process Tube Elongation Forces," A. Russell, December 5, 1973.

A nominal clearance between adjacent front connectors of 7/8 inch was established at lay-up. Differential tube elongation rates alter these clearances as tube length changes are realized by relative tube displacements with respect to the front hardware tube expansion assembly. An examination of the front connectors revealed two locations where connector clearances at the elbow, where the connectors depart from their front-to-rear run to the side-to-side run towards the risers, have been reduced to 1/8-1/16 inch. The connectors are rigid in this area because the front-to-rear piping runs are short and the connectors are securely fixed to the nozzles by the Grayloc connectors. Thus, adjacent connectors bearing against each other would significantly increase member stresses as tube elongation progressed at different rates. This problem is under active consideration to develop corrective capabilities as necessary for the isolated cases which will require near-term action.

Primary Loop Debris

Primary loop flushes of the dead leg at the base of each inlet riser continue to be conducted at each scheduled outage. The flushes are collected in a common drain and filtered (0.030 mesh screen) to remove debris. Collected debris remains relative consistent in terms of quantities, 5-22 cc, and constituents, roughly equal volumetric portions of sand-like material and discrete 1/4 to 1/2 inch major dimension metallic fragments identified as Stellite 6B.¹

The collected debris measures from 0 to >500 R/hr at approximately one foot in air. The flushes typically reduce dose rates at the base of the inlet risers by a factor of 10-15, indicating the flushes are efficient in removing most of the debris. The flushes which are conducted prior to the forward flow tests of the CV-2R valves will continue to assure that the potential for reintroducing such debris into the loop in the course of performing the required ECS system testing is minimized.

Surveillance Activities

In 1973, a draft of a formal Technical Division surveillance program was prepared.² The program identifies inspection of materials and equipment not currently covered by other programs, such as Equipment Maintenance Standards or the Pressure Systems Instructions, and whose unanticipated failure could significantly affect the nuclear safety and/or operability status of the plant. The program has been reviewed and factored into outage planning. The document will be issued in 1974 and updated annually.

1. DUN-8112

2. UNI-70, "N Reactor Materials Surveillance Program," Technical Division Staff, Issued by N. R. Miller, September 14, 1973.

CY-1973 Special Surveillance accomplishments are as follows:

- Steam Generator Tubes

Approximately 140 Inconel-600 steam generator tubes were eddy-current tested in steam generator 4B in May 1973, and 85 304-stainless steel tubes were similarly tested in steam generator 5B during December 1973. The testing did not indicate any significant deterioration of the steam generator tubing had occurred.

Tube 4AS R19L1 was removed from steam generator 4A in May 1973 for destructive examination. Preliminary results have not revealed any significant corrosion attack indicative of general tube deterioration. The examination has not been completed due to other priorities.

- Rod Cooling System

A right side control rod cooling system heat exchanger tube bundle was visually examined during the summer 1973 outage. No evidence of accelerated corrosion of the Admiralty Metal tubes was detected.

- Titanium Sheaths

The procedure for replacement of the control rod system poison inserts during the summer 1973 outage (see page 23) included a requirement to hydrostatically test each control rod tip section titanium sheath at 250 psig. Two sheaths (from HCR channels 13 and 58) failed this hydrostatic test. Both sheaths, based on visual examinations made during the tests, were reported to leak in the area of the circumferential weld that joins the bullet-nose tip to the tubular portion of the sheath. The leak in sheath 58 was confirmed by subsequent inspections; the leak in sheath 13 has not been verified even though testing included a hydrostatic test at 450 psig, x-radiography, and dye penetrant inspections. Destructive examinations are in progress on both sheaths. Metallographic examinations so far reveal extremely large grain sizes in the weld areas, which is indicative of poor overall weld quality. In addition, the crack surfaces on sheath 58 exhibit characteristics of a brittle-type failure. Hydrogen extractions showed that the levels had not increased above the base content, however. The analytical portion of the evaluation required to assess these findings in terms of actual cause of failure and possible implications on the remaining sheaths has not been completed.

- Grayloc Stud Bolt Evaluation

Twenty Grayloc stud bolts were removed for destructive examination during the 1973 summer outage V-12 valve repair efforts.¹ V-12 valve Grayloc studs are selected for these evaluations because of their severe service environment. Evaluations to date have shown no statistically significant decrease in the strength or ductility of the bolts. In addition, the data show that no deleterious effects resulted from the external rear pipe space piping decontaminations conducted in 1972.

1. UNI-72, "1973 Grayloc Stud Bolt Evaluation," W. K. Kratzer, September 17, 1973.

TESTING

Confinement System

The confiner leak rate measured during the summer 1973 outage was 3.0 percent, and subsequent to the test measurement, leaks were repaired in the confiner exhaust fan ducts. The leak rate was therefore considered to be below the three percent required by Equipment Maintenance Standards to permit a year's operation prior to another test. The leak rate was 1.5 percent in 1972 and 2.5 percent in 1971. These rates compare favorably with the five percent leakage used in accident calculations. The integrated confinement and fog spray tests were satisfactorily completed with only minor difficulties.

Halogen Filters

The Zone I halogen filters are subjected to semiannual in-place iodine retention tests to assure their ability to adequately perform under accident conditions. Results from these charcoal efficiency tests are summarized below.

ZONE I HALOGEN FILTER TEST RESULTS - HOT EXTRAPOLATED ACCIDENT EFFICIENCIES*

<u>Cell</u>	<u>Percent Efficiency Previous CY-1972 Test</u>	<u>Percent Efficiency Last CY-1973 Test**</u>
A	99.07 (12/72)	99.53 (6/73)
B	99.77 (6/72)	99.58 (6/73)
D	99.79 (6/72)	99.02 (10/73)

The low efficiency measured in December 1972 for A cell reflects the degree of variability that is unexplainably noted from time to time in the tests. All extrapolated accident efficiencies, however, remain well in excess of the required 95 percent Process Standard limit. Efforts to determine charcoal replacement dates are scheduled for completion about mid-calendar year.

* The "hot extrapolated accident efficiency" is a measure of filter performance under actual accident conditions. The results of an ambient temperature laboratory efficiency test are subtracted from those of an ambient in-place test to establish a cell leakage rate. This leakage is then subtracted from the results of a laboratory test on actual cell charcoal samples conducted at accident conditions (100 C, 100 percent relative humidity) to arrive at the hot extrapolated accident efficiency for a given cell.

**December 1973 results not yet available, but preliminary results indicate no significant loss in efficiency.

Particulate Filters

Semiannual tests of the particulate (HEPA) filters are required. The tests are conducted by removing artificially produced 0.3 micron diameter di-octyl pthalate (DOP) from the air stream at normal air flow conditions. The results of CY-1973 testing is summarized below:

DOP TEST PARTICULATE FILTER EFFICIENCIES

<u>Test Date</u>	<u>3/14/73</u>	<u>6/1/73</u>	<u>7/5/73</u>	<u>10/8/73</u>	<u>12/12/73</u>
<u>Cell</u>					
A		99.956	99.985*	99.982	
B		99.956	99.986	99.987	
C	99.988			<90.000	99.996*
D	99.992			99.996	

*New Filters

The low particulate filter efficiencies obtained on A and B cells on June 1, 1973, are not fully understood. The A cell particulates were replaced with new filters on July 2, 1973, to assure that the 99.95 filtration efficiency Process Standard limit would be met for the Zone I (A and B cells) effluent air stream. On two subsequent tests in CY-1973, the B cell particulates showed satisfactory results.

The extremely low efficiency of C cell particulate filters recorded on October 8, 1973, is discussed under Unusual Occurrences, page 52.

A-B Electrical Bus Hydraulic Interdependence Test, PT-NR-125

The PT-NR-125 testing program¹⁻⁶ was initiated in 1969 to identify the cause of an observed hydraulic interdependence between the A- and B-Bus electrical

1. DUN-6181, "PT-NR-125, A-B Electrical Bus Hydraulic Interdependence Test," J. L. Tenwinkel, August 21, 1969.
2. DUN-6181, SUP 1, same title, J. L. Tenwinkel, August 29, 1969.
3. DUN-6181, SUP 2, same title, J. L. Tenwinkel, December 2, 1969.
4. DUN-6181, SUP 3, "PT-NR-125, A-B Electrical Bus Hydraulic Interdependence Test - CRW System Cycling," J. L. Tenwinkel, February 25, 1970.
5. DUN-6181, SUP 4, "PT-NR-125, A-B Electrical Bus Hydraulic Interdependence Test," J. L. Tenwinkel, July 9, 1970.
6. DUN-6181, SUP 5, same title, R. E. Baars and K. E. Fields, November 22, 1972.

systems. The results of the initial test¹ and those performed under Supplements 1, 2, and 3, indicated that the turbine generator (TG) condenser instability problem was related to leakage of air into the system. During testing conducted under Supplement 4, it was noted that a large amount of air was being introduced into the circulating raw water (CRW) system condenser discharge line from the vacuum priming pump lines.² This air subsequently formed a large bubble at the CRW outlet of the surface condenser and caused a loss of syphon. Based on these results, an operating mode was established where the vacuum priming pump lines were isolated from the surface condenser and the CRW system was valved to supply a large enough flow (~25,000 gpm) to the condenser upon loss of A Bus to sweep air from the system. The increased CRW flow to the surface condenser was provided by reducing flow to the dump condensers which reduces plant thermal power level capability when steam is not being exported for electrical power generation. During use of this operating mode, during the first half of 1973 there were no indication of surface condenser instability.

Further incentives to study TG set operation resulted from a request³ to establish the modifications necessary for reactor operation in the plutonium-only mode at 4000 Mwt. Resolution of the TG set instability problem would permit a reduction in the CRW flow rate to the TG set surface condenser, thereby permitting a more efficient use of the CRW system cooling capacity and allowing a 4000 Mwt year-round reactor operation with no steam export for electrical generation.

Testing to accommodate this request was authorized by PT-NR-125, Supplement 5. The primary objective of this test was to determine if an adequate air extraction system (AE) on the CRW side of the turbine generator surface condenser would improve TG set reliability and permit returning CRW flow to the condenser to near-design flow rates. The testing was conducted with an oversized AE system in operation and CRW flow rates to the condenser of 8,000 to 20,000 gpm for the cases of no load on the turbine, and for an 11 Mwe load on the turbine. To evaluate the effect of air in-leakage on the TG set operating stability, air was purposely injected at rates of 0 to 56 cfm. The test results indicated that the TG surface condenser could be operated with a reduced CRW flow rate provided an adequate AE system was in operation.

Based on these results, Design Change 3239⁵ was initiated to provide an adequate system to serve the CRW side of the TG set surface condenser. The modifications

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1. DUN-6373, "PT-NR-125 and PT-NR-125, Supplement 1, Interim Report - A-B Bus Interdependence Test," J. L. Tenwinkel and D. L. Renberger, October 27, 1969.
 2. Letter, J. L. Tenwinkel to Distribution, "Interim Summary Report, PT-NR-125, Supplement 4," January 4, 1971.
 3. Letter, O. J. Elgert to C. D. Harrington, "N Reactor Operation - Plutonium-Only Mode," dated July 10, 1972.
 4. DUN-6181 B, "PT-NR-125, Supplement 5, A-B Electrical Bus Hydraulic Interdependence Test, Final Report," R. E. Baars and K. E. Fields, January 22, 1973.
 5. Design Change 3239, "Turbine Generator Condenser Coolant Side Primary and Extraction System," C. W. Harrison, March 1, 1973.

and installation of equipment authorized by Design Change 3239 were completed in July 1973. PT-NR-125, Supplement 6¹, was prepared to serve as an Operational Test Procedure (OTP) for the new air extraction system. The results of the Supplement 6 tests completed during the 1973 summer outage and their application to operation of the TG set condenser is summarized below. A complete report of the test results is documented as Reference 2. The objectives of this test were as follows:

1. To demonstrate that the turbine generator can be operated with a high degree of reliability at near-design coolant flow rates with the new AE system in operation.
2. To collect data for a Technical Basis defining operation of the TG surface condenser at near-design coolant flow rates when the AE system is in operation.
3. To obtain information on the CRW system flow distribution at near-design TG condenser coolant flow rates.

All objectives were achieved. The minimum flow rate for stable operation of the condenser at an inlet coolant temperature of 67 F was determined to be 10,000 gpm. With the condenser valved for a flow rate of 12,000 gpm on two-pump operation, a simulated loss of A Bus power from four-pump operation was survived with no indication of instability.

A loss of syphon was induced by deactivating the AE system while injecting air at a rate of 60 cfm. A turbine low-vacuum alarm was received at once, closely followed by turbine trip-off. This test demonstrated the need for a reliable air extraction system when operating at design flow rates.

Thus, the test results confirm that the TG set can be reliably operated at near-design flow rates when the air extraction system is in operation. Reliability of the AE system is critical; without the system in operation, there would be an increased probability of loss of syphon and TG trip-off upon a loss of A Bus power. The test data also reconfirmed the adequacy of the previous limits in assuring reliable turbine operation under circumstances where no air extraction system exists.

A revised limit structure based on the test results were developed and incorporated into the Process Standards. The new minimum condenser coolant flow rate limits provide at least a 25 percent safety margin compared to test results. It should be noted that these limits provide greater than 25 percent safety margin during months when the river temperature is below 68 F. If river temperatures exceed 68 F, procedures are provided to determine the corresponding condenser flow rate limit.

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1. DUN-6181 C, "PT-NR-125, Supplement 6, "A-B Electrical Bus Hydraulic Interdependence Test OTP for Air Extraction System (Design Change 3239)," R. E. Baars and K. E. Fields, April 20, 1973.
 2. DUN-6181 D, "PT-NR-125, Supplement 6, "Final Report, A-B Electrical Bus Hydraulic Interdependence Test OTP for Air Extraction System (Design Change 3239)," R. E. Baars and K. E. Fields, November 14, 1973.

The new limits with the air extraction system operable are compared in Table I to previous limits. Previously, stable operation of the condenser was achieved by specifying sufficient inlet pressure to assure sufficient flow even if syphon action were lost due to air buildup in the system. The previous flow rate limits corresponding to these pressures are shown in Table I under "AE System Inoperable."

Data collected on the CRW system flow distribution with the TG condenser valved as per these revised limits is shown in Table II. It is concluded that N Reactor will have a 4000 Mw power level capability year-round with no steam export to the Washington Public Power Supply System steam plant, with the air extraction system operable.

TABLE I
CONDENSER CRW SYSTEM FLOW RATE LIMITS

<u>Pumps Operating</u>	<u>AE System Operable</u>	<u>AE System Inoperable</u>
2 on B Bus	15,000 gpm	25,000 gpm
1 on A Bus 2 on B Bus	20,000 gpm	29,000 gpm
2 on A Bus 2 on B Bus	24,000 gpm	33,000 gpm

N Reactor Rupture Monitor Flow Determination - PT-N-241¹

One of the bleed flows used for determining primary loop leak analysis is the flow through the rupture monitor system. Since no installed instrumentation was available for determining this flow directly, PT-N-241 was designed to permit periodic determination of rupture monitor system flow during reactor operation. The procedure for obtaining the rupture monitor system flow consisted of measuring the difference between high pressure injection flow before and after valving the rupture monitor system out of service.

Repeated measurements utilizing PT-N-241 has shown that total primary coolant flow through the rupture monitor system has not varied significantly over the past several years. Since inception of PT-N-241, five flow determination tests were performed. Data from these tests show that the total flow through the rupture monitor system averages 230 gpm. Consistency of the data also suggests a reasonable degree of accuracy, certainly within the error of high pressure injection flow measurement instrumentation. Based upon the consistent flow data obtained, PT-N-241 was terminated.

1. DUN-7308, "PT-NR-241, N Reactor Rupture Monitor Flow Determination," R. J. Nicklas, January 5, 1971.

TABLE IICRW SYSTEM FLOW DISTRIBUTION

As measured with reactor
down, system unthrottled

<u>CWR Pump Configuration</u>	<u>2 Pump - gpm</u>	<u>3 Pump - gpm</u>	<u>4 Pump - gpm</u>
Dump Condensers	166,100	225,150	274,500
DT Condensers	32,550	42,350	52,000
TG Condenser	15,000	20,000	24,000
GCS Heat Exchangers	7,700	10,000	12,000*
182 Building (Est)	10,000	10,000	10,000
Filter Plant (Est)	6,000	6,000	6,000
Total Equipment Flow	237,350 gpm	313,500 gpm	378,500 gpm
Total CRW Pump Flow	242,200 gpm	330,000 gpm**	391,500 gpm
Difference: (Total Pump Minus Total Equipment)	4,650 gpm	16,500 gpm	13,000 gpm

* Extrapolated from two-pump case flow.

** Data were inadvertently missed on No. 3 pump;
for this case, No. 3 pump flow was assumed
equal to No. 2 pump flow.

Reactor Coolant Temperature Measurement with Zero Primary Flow to Establish Maximum Flow Shutoff Time - PT-N-331¹

The purpose of Production Test N-331 was to measure maximum flow shutoff time for the reactor conditions (fuel exposure, downtime, operating history) that existed at the time of the test. If the maximum flow shutoff time was measured to be long enough, it was planned to use the "flow stagnation method" for isolating the CV-3 and CV-5 valves for repair during the summer outage. The test was repeated twice during the outage, 20 days after shutdown, and again 42 days after shutdown. For both tests the flow shutoff time was found to be too short to be usable for valve repair. Fuel temperatures were monitored throughout the test by special thermocouple probes that had been inserted into the center of high power and high exposure fuel columns. The tests were terminated when the maximum fuel temperature reached 200 F. The capability to restore full shutdown recirculation cooling was retained at all times during the test, and fuel temperatures were continuously monitored and recorded.

A Procedure for Establishing Once-Through Coolant Flow During Major Reactor Component Maintenance - PT-N-341²

The purpose of Production Test N-341 was to develop a cooling scheme that would allow repair of the CV-3 and CV-5 check valves which could not be isolated with the primary system on recirculation operation, and then to actually use the method to complete the repairs during the 1973 summer outage. A successful cooling scheme was developed that used a once-through demineralized water coolant flow rather than the normal recirculation cooling. The once-through flow path allowed draining the primary loop piping in the vicinity of the check valves so that the repair work could be completed. Throughout the testing and repair work, the capability was retained to return to the normal recirculation cooling mode. It is anticipated that this cooling mode may again be used during the 1974 summer outage for additional primary loop inspection and valve repair.

Reactivity Increases with Central Zone Spike - PT-N-344³

Operating experience to date with Mark IV fuel and analytical predictions have indicated that the increase in the fuel exposures for a product assay near 12 wt% Pu-240 requires additional excess reactivity at the beginning of any given operating period to maintain an adequate level of excess reactivity (approximately one milli-k or greater) at the end of the run. Accordingly, a production test is in progress to determine the effectiveness of a scheme to increase reactivity by charging spike (Mark IA) fuel in the central zone within the enrichment ring.

1. DUN-8141, "PT-N-331, Reactor Coolant Temperature Measurement with Zero Primary Flow to Establish Maximum Flow Shutoff Time," K. N. Cummings, April 19, 1973.
2. UNI-26, "PT-N-341, A Procedure for Establishing Low Once-Through Coolant Flow During Major Reactor Component Maintenance," D. W. Whitcomb, June 7, 1973.
3. UNI-75, "PT-N-344, Addition of Mark IA Fuel to Central Zone of N Reactor," B. W. Howes, October 2, 1973.

The purpose of the production test is to obtain enthalpy imbalance measurements for both the 10-piece and 15-piece Mark IA test columns irradiated for an average fuel exposure of 2450 Mwd/ton and to determine what limitations to reactor power would correspond to the gains in excess reactivity that would be achieved by central zone charging of spike columns. An additional goal is that of determining the number of spike columns that would be required to reach the product assay goal.

Excess reactivity decreases by about 2.5 mk during a 56-day operating period at full power in a product mode approaching 12 percent, and extra Mark IA spike fuel columns are being charged each outage to gain the reactivity required to operate in the planned mode. The total number of central zone Mark IA fuel columns required to operate N Reactor in this production mode has been estimated to be 50. The reactivity increases per exchange have been higher than expected, and the operating parameters such as tube powers and outlet temperatures have remained within operating limits at full power (3770 Mwt). Thus, all indications are that the method for increasing reactivity will be successful.

N Reactor Rod Calibration - PT-N-336¹

The rod calibration test performed following replacement of the control rod tip is described on page 24.

Summary of N Reactor Production and Development Test Status - CY-1973

<u>Type</u>	<u>Active 12-15-72</u>	<u>Approved 1973</u>	<u>Completed 1973</u>	<u>Active 12-31-73</u>	<u>Final Reports 1973</u>
Equipment	6	7	5	6	6
Fuels & Materials	11	5	5	7	9
Instruments	6	6	2	8	5
Operational	9	9	6	3	7
TOTAL	32	27	18	24	27

1. UNI-13, "PT-N-336, N Reactor Rod Calibration," D. L. Larkin, June 6, 1973.

- CHANGES IN COMPANY ORGANIZATION STRUCTURE

Effective April 4, 1973, the United Nuclear Corporation purchased McDonnell Douglas Corporation's interest in Douglas United Nuclear, Inc. The new company, 100 percent owned by United Nuclear Corporation, is named United Nuclear Industries, Inc. There were no changes of company management at Hanford as a result of the purchase.

SUMMARY OF N REACTOR OUTAGES AND OUTAGE EXTENSIONS - CY 1973

<u>Shutdown Date/Time</u>		<u>Startup Date/Time</u>		<u>Duration Hours</u>	<u>Type</u>	<u>Reason for Shutdown</u>
1-26	2040	1-28	2110	48.5	Scheduled	Charge-Discharge (90 tubes)
1-29	0051	1-29	0214	1.4	Manual	To retrieve dropped balls from hopper 38 gate failure
1-29	0951	1-30	1033	24.7	Manual	To repair primary coolant leak in Cell 6
2-16	1703	2-17	1650	23.8	Scram	Drop of Ball Hopper 61
3-2	2046	3-11	2010	215.4	Scheduled	Charge-Discharge & Maintenance
4-2	1653	4-3	1525	22.5	Manual	Fuel Failure - Tube 0557
4-9	2208	4-10	2010	22.1	Manual	To check out abnormal flow indications in rupture monitor sample lines
5-12	1110	8-7	2245	2099.5	Scheduled	Summer Maintenance
* 8-8	0920	8-8	1202	2.7	Manual	Seat V-7 valve in primary loop
* 8-8	1334	8-8	1420	0.8	Scram	Spurious flow monitor trip
* 8-8	1731	8-8	2236	5.1	Manual	Check out diversion header leakage
* 8-9	0656	8-9	1026	3.5	Scram	Secondary system low surge tank level
* 8-10	1520	8-11	0748	16.5	Scram	Spurious flow monitor trip
* 8-11	1540	8-11	2049	5.2	Manual	Correct graphite cooling system problems
8-30	0415	8-31	1058	30.7	Manual	Fuel Failure - Tube 1652
** 8-31	1111	8-31	1210	1.0	Scram	Intermediate flux monitor trip due to bypass error
9-9	1336	9-10	2131	31.9	Scram	Loss of #6 drive turbine
10-1	0340	10-2	0622	26.7	Scram	Flow monitor toggle valve leak
10-5	2018	10-14	2035	216.3	Scheduled	Charge-discharge and Maintenance
** 10-14	2308	10-15	0548	6.7	Manual	RTD repair on disturbed tube
** 10-15	0848	10-16	0001	15.2	Scram	Loss of #1 drive turbine vacuum
** 10-16	2340	10-17	2351	24.2	Manual	Repair leaking V-12 valve and faulty RTD
10-22	0654	10-23	0415	21.4	Scram	Fuel Failure - Tube 3364
12-5	0029	12-6	0328	27.0	Scram	Spurious flow monitor trip
** 12-6	0528	12-6	1127	6.0	Scram	Spurious flow monitor trip
12-7	0207	12-8	2052	42.7	Scram	Fuel rupture indications
12-21	2258	12-31	2400 (Yr.End)	241.0	Scheduled	Charge-Discharge & Maintenance

*Theses shutdowns occurred during the plant check-out program following summer outage.

**These shutdowns occurred during reactor start-up.

MAINTENANCE

Leak Repairs - Primary Coolant System

Primary coolant system makeup flow requirements in January 1973 were 940 gpm. Approximately 228 gpm of this amount was attributed to leakage (valve packing leaks, valve leak-through into the vent and drain systems, and other leakage).

At year end the primary coolant system makeup flow requirements were 1200 gpm, of which approximately 439 gpm was due to leakage. Following the December 1973 outage, which continued across year end, the coolant system makeup flow requirements were 1180 gpm, of which approximately 285 gpm was due to leakage.

The leak repairs accomplished during CY-1973 were:

Miscellaneous Valves and Connections	58
V-12 Valves Replaced in Kind	11
V-12 Valves Replaced by Spool Pieces	62
V-11 Valves Replaced or Hub Leak Repaired	3
MUS Leakage Corrected	7
Other Grayloc Connector Leaks Repaired	4
V-11 Valves Repacked	4

CV-3 and PCSV-202 Valve Repair

During May 1973, the primary side of steam generator 4A was opened for inspection resulting in the discovery of debris consisting of about three dozen pieces of metallic fragments. Subsequent inspection of steam generator 4B led to a similar discovery. The debris was removed, and most was identified as pieces of Stellite bushings used in the primary loop CV-3, CV-5, and PCSV-202 valves (see page 46). An inspection and repair program was initiated during the 1973 summer maintenance outage which resulted in the installation of new bushings and shafts comparable to the original installation in the 16 CV-3 valves and the six PCSV-202 valves. After three of the 16 CV-5 valves were inspected and found to be in sound condition, further inspection of CV-5 valves was deemed unnecessary.

Safety Rod Tip Section Replacement

Procurement and Replacement of Rod Tip Sections - As a result of continuing study and monitoring of rod component integrity, the decision was made in early 1973¹ to replace tip inserts in all horizontal rods during the summer 1973 outage. Procurement of a full complement of tip inserts and lesser quantities of other components essential to complete refurbishing was accomplished to permit replacement of tip inserts and other components as required for 83 of the 85 active rod positions.

1. DUN-8125, "Douglas United Nuclear Monthly Report," February 1973.

Stringent quality assurance^{1,2} for procurement and replacement, subsequent nuclear calibrations³, and improved control of pitting corrosion via the recirculatory mode of cooling, assure that the corrosion problem⁴ and the attendant question of nuclear integrity associated with the rod tip sections have been resolved for the life of the plant.

Validation of Replacement Safety Rod Tip Sections - Neutron absorption characteristics tests using epithermal neutrons generated by Cf-252 were successfully completed for all of the replacement HCRs. The tests demonstrated the presence of B₄C, measured density variations along the length of the control rod tips, and showed that there were no void regions of significant size in the control rod cavities. Similar measurements on an original HCR tip provided a comparison of the neutron absorption characteristics between the replacement and original HCR tips. The difference in B₄C density between the old HCR (1.8 g/cc) and the replacement HCR (1.6 - 1.5 g/cc) was compensated for by larger size B₄C channels in the replacement rods resulting in nearly identical measured neutron absorption characteristics for the two sets of rods.

During the course of the experiments, count rates for all three channels of a control rod tip were recorded as a continuous trace on strip chart recorders for immediate evaluation, and as integrated digital data on paper tape for later analysis. The digital information was combined with HCR tip dimensional data and weights of B₄C loadings in a special computer program. This computer program corrected the count rate data for the decay of the Cf-252 source, established a correlation between measured densities and count rates, and assigned an average density value to each B₄C channel with error limits indicative of density variations along the length of the control rod tip. The results of this analysis for each rod was included in the data file on each rod.

Rod Calibration Test - Following replacement of the rod tips, physics tests were conducted and rod strengths were determined for several rod patterns in order to verify predicted rod worths. Individual rod worths were measured using the free rising period technique. Following step insertions of individual control rods, period data were obtained from seven independent neutron channels located in different regions of the reactor. Individual rod worths were determined from the average of the free rising period measurements. These data are shown for comparison with previously obtained rod worth measurements in Figure 1. Discrepancies, where they occur, between old and new individual rod calibration data are due largely to changes which have occurred in the neutron flux distribution as a result of the addition of the spike ring and residual samarium balls in the stack. As indicated in Figure 2, the new rod system strength exceeds the old data by six percent which is in excellent agreement

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1. UNI-8, "United Nuclear Industries Monthly Report," May 1973.
 2. UNI-47, "United Nuclear Industries Monthly Report," July 1973.
 3. UNI-48, "United Nuclear Industries Monthly Report," August 1974.
 4. DUN-8112, "Annual Report of N Reactor Operating Experience Pertinent to Nuclear Safety - CY-1972," Staff of Douglas United Nuclear, February 15, 1973.

FIGURE 2

		1.099 2.11	.134 .13	.166 .14		.206 .14	.206 .14	.204 .12			9
	.203 .23	.324 .29	.491 .34	.644 .39	.578 .41	.788 .40	.744 .38	.623 .34	.505 .28		8
.173 .30	.324 .41	.495 .57	.784 .64	1.035 .73	1.006 .77	1.329 .76	1.217 .71	.964 .62	.752 .52	.453 .40	7
.203 .41	.377 .56	.587 .73	.984 .91	1.359 1.04	1.264 1.08	1.711 1.08	1.499 1.00	1.127 .87	.874 .72	.520 .55	6
.189 .45	.363 .63		.961 1.02	1.326 1.19	1.267 1.25	1.621 1.24	1.430 1.15	1.068 .99	.820 .82	.497 .64	5
.165 .42	.305 .58	.475 .77	.799 .95	1.136 1.10	1.055 1.18	1.373 1.17		.871 .94	.666 .78	.414 .61	4
.129 .32	.231 .44	.353 .61	.574 .70	.782 .80	.730 .86	.903 .86	.801 .80	.628 .71	.497 .59	.306 .46	3
	.146 .25	.221 .32	.340 .39	.451 .44	.410 .47	.514 .47	.472 .44	.386 .39	.312 .33		2
		.059 .12	.085 .14	.093 .17	.091 .19	.126 .18	.221 .16	.124 .14			1
0	1	2	3	4	5	6	7	8	9	10	

¹Top - Individual Rod Strengths (mk) Total = 51.738.

²Bottom - Expected Rod Strength (mk) from HW-82100 PT3, Startup Physics Test,
Total = 48.85.

with the calculated increase. This increase is due primarily to the addition of the spike ring which tends to flatten the neutron flux distribution from side to side.

Figures 3 and 4 compare the relative rod strengths with the square of the neutron flux distributions obtained during the test. Since the relative rod strengths are expected to be roughly proportional to the square of the neutron flux, these comparisons demonstrate that the individual control rod strengths do indeed follow the flux-squared distribution which indicates that the measurements are correct. The one notable exception is Rod 71 which has a measured worth that is approximately 80 percent higher than expected. The reason for this is being investigated.

Following the individual rod calibration test, approximately one-half of the operational rod withdrawal sequence, Pattern D, was calibrated. This calibration provides an accurate determination of reactor excess reactivity and an approximate determination of total system worth that will reduce the uncertainty in critical predictions and long-term reactivity projections. The results obtained during this portion of the test agree reasonably well with the values in current usage with the exception of rod pairs 104/16 and 106/14. The latest values for these two rod pairs are approximately 25 percent of the currently accepted values. There is good reason to believe that the latest measurements are more correct and should be incorporated in the new Pattern D rod worths.

The final portion of the test consisted of critical rod pulls for rod withdrawal sequences which would demonstrate the effects of removing groups or clusters of rods from one area of the reactor core. These tests also furnished information on the magnitude of reactivity ramps used in kinetic studies and speed-of-control. The results of the hazards configuration rod patterns seem to indicate that the conditions tested are not as hazardous as previously assumed. Since the earlier rod calibration data yielded a lower rod strength value than the present data seems to indicate, current usage of the previous calibration data is conservative.

In addition to verifying the strength of the new rod tip inserts, a secondary objective of the rod calibration test was to determine individual rod worths as well as total system worth in the presence of residual samarium oxide balls which escaped in the reactor stack through cracks in the ball system channel liners. The residual balls are in competition with the central rods for absorption of neutrons and could effect rod worths. It is concluded with the calibration test results that the presence of the residual samarium oxide balls has not had a large impact on rod worths.

Rehabilitation of Top Row HCR Channels

No active on-reactor work has as yet been accomplished which will result in rehabilitation of the top row HCR channels. As a result, HCR No. 59 is being maintained out of service, channels Nos. 49 and 69 are in service but not scrammable, and channels Nos. 29, 39, 79, and 89 are in normal service. (See page 9).

• + 71

FIGURE 3
RELATIVE ROD STRENGTHS
HORIZONTAL DIRECTION

FIGURE 4
RELATIVE ROD STRENGTHS
VERTICAL DIRECTION

Rod Row O

Rupture Monitor System Upgrade

As reported in last year's report, plugging of the raw water cooling side of the 1003 rupture monitor sample line heat exchangers is a continuing problem. The program of heat exchanger replacement was continued. Severe degradation and periodic failure of the raw water strainers resulted in an increased rate of heat exchanger failure. Replacement strainers were procured and installed during the summer of 1973. Installation of the new strainers has decreased the rate of heat exchanger failure, but some units that were plugged or partially plugged prior to strainer replacement will be replaced in 1974.

During the summer 1973 outage a complete overhaul of the 3009 toggle valves and elimination of 2006 toggle valves by replacing them with plugs was accomplished.

Corrosion studies indicate the possibility that appropriate chemical additions may be helpful in decreasing the rate of boiler scale type deposits in the raw water system and reducing the rate of heat exchanger plugging. Equipment additions for chemical addition has been designed, fabricated, and installed. Work towards placing the system in operation is underway.

Studies of the electronics of the rupture monitor system indicate that a more reliable system would result if a log ratio system was installed in place of the current linear difference system. The benefits of the log ratio system are:

- Wide variation in sample activity will not vary the calibration of the instrument.
- Wider variation in effluent sample delay times will be tolerated.
- Monitoring coverage of effluent sample will be obtained from startup through a rupture indication without range changing.

A program to convert one turret (92 sample points) as a prototypic installation is planned for calendar year 1974.

Increased Sensitivity Bulk Rupture Monitor

A 109-N primary coolant gross gamma rupture monitor system was placed in service during April 1973 and subsequently a recorder and associated annunciation was added in the 105-N control room. The system is now operating to provide backup to the primary rupture monitor system.

A second backup rupture monitor (iodine monitor) was also installed in calendar year 1973. This monitor samples the primary system coolant after it leaves the steam generator. Solvent extraction is used to separate the iodine fraction of the radionuclides to obtain a background - free iodine spectrum for gamma spectrometry. The primary purpose of the monitor is to

assist in detecting and confirming fuel failures; however, it is anticipated that it will also provide a measure of the primary coolant system radioiodine inventory needed for monitoring environmental releases. Evaluation of the capabilities and limitations of this instrumentation is in progress.

Rod Recirculation System Repair

The rotating assembly for the No. 2 right-side recirculation pump failed, resulting in a system dump and single-pass operation. The failure resulted from high stresses caused by improperly aligned suction pipe. Expansion joints have been installed at the suction side of all pumps and pump-to-motor alignment has been checked to prevent a recurrence of this type of failure. (See page 53).

PROCESS STANDARDS AND EQUIPMENT MAINTENANCE STANDARDS COMPLIANCE

Process Standards Audit Activities

There were 11 violations of Process Standards reported for CY-1973, as compared to 23 violations during CY-1972. During CY-1972, 73 percent of one engineer's time was spent in auditing and audit-related activities compared to 39 percent of one engineer's time for CY-1973. However, auditing time has remained essentially constant for CY-1972 and CY-1973, whereas time for audit-related activities, such as preparation of checklists, has decreased.

Audits of Process Standards requirements and production test special limits are made by Process Evaluation Unit, Process Application Unit, and W Plant management. Percentiles shown above only include the audit effort by the Process Evaluation Unit. A brief listing of the violations reported by these sources is given below.

February 10	Annunciator power was lost to annunciator panels 1, 2, 3, and 4 in the 105 Control Room. (See page 44.)
March 5	Five rods were out of service during a reactor outage which violated the three rod out-of-service limitation of Standards. (See page 45.)
March 11 & 12	Rupture monitor sample block valves were closed during reactor operation at 500 Mw. (See page 45.)
May 5	Green circuit CD-16 (125 v dc) was found de-energized during reactor operation. This circuit is one of two sources of backup power to annunciator panels for Cell 6. (See page 46.)
August 12	The surge tank for the graphite cooling system was full of water as a result of level control system failure. Manual level control was authorized. (See page 49.)
September 10	A drain valve, V-57, for the pilot valve for RV-2-1 primary loop relief valve was closed. This resulted in RV-2-1 being out of service. The second relief valve, RV-2-2, was in service and alone has adequate capacity. (See page 51.)
September 21	Gas activity of the reactor gas atmosphere was out of limits as a result of failure of the automatic gas makeup system. (See page 51.)
October 24	Confinement door circuit CE-20 was found energized by a jumper cord during reactor operation. (See page 52.)

- November 20 Main steam header pressure to condensate pump suction header ΔP exceeded the 3 psi limit during EMS testing of equipment. (See page 53.)
- December 6 The rupture monitor was found valved off process tube 2350 which contained a ruptured fuel element. (See pages 53-55.)
- December 21 Main steam header pressure to condensate pump suction header ΔP exceeded the 3 psi limit while returning dump condenser No. 14 to service.

EMS Program and Audit Activities

Refinement and updating Equipment Maintenance Standards to simplify implementation and assure current procedures continued. Steps were initiated to provide the EMS Unit full control of EMS-related field work sheets and to specifically identify critical inspections requiring special attention, thus improving audit visibility in these areas.

The quarterly 100 percent audit of EMS data sheets was continued throughout the year, supplemented by witness audits. The Critical Systems Data File was successfully utilized for identification of compliance discrepancies. Refinement and updating of the Critical Systems Data File also continued, concurrent with EMS updating. Audit results for the period reflected the expected level of compliance performance.

NUCLEAR SAFETY ADMINISTRATION

Operating Safety Limits

The new AEC Manual Chapter 0540 (June 20, 1972) requires that technical specifications replace Operating Safety Limits (OSLs) for all AEC-operated reactors now in service. In August 1972, an agreement was reached with AEC-RL, that because of the limited projected operating span of N Reactor, an updating of the OSLs¹ and provisions of technical bases for Section V of the OSLs would satisfy the intent of Manual Chapter requirements². The revised OSLs, which incorporate technical bases for Section V, were transmitted to AEC-RL in Reference 3. The revised OSL limits with bases for Section V have been approved by AEC-RL⁴ and incorporated into UNI administrative practices.

Technical Specifications

As noted in the previous paragraph, AEC Manual Chapter 0540 requires that field office managers assure that current technical specifications exist for all AEC-owned reactors under their surveillance. A review of this requirement by AEC-RL⁵ and United Nuclear Industries staff concluded that technical specifications would be required for N Reactor and would replace the existing Operating Safety Limits. A UNI Technical Specifications Program was begun in March 1973 in anticipation of an AEC-RL request to scope and summarize the requirements of initiating a Technical Specification program for N Reactor. The scope and summary was requested⁶, transmitted to⁷, and subsequently used by AEC-RL in preparing a request⁸ that UNI initiate a program to prepare technical specifications for N Reactor. In agreement with the request, UNI established and initiated a program to prepare technical specifications for N Reactor on a schedule that will permit completion of approved specifications by October 1974 and full implementation by January 1975.

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1. RL-OSL-NPR (DUN-I-47), "Operating Safety Limits N Reactor," AEC-RL, December 5, 1967.
 2. Letter, R. Ritchie to C. D. Harrington, "Operating Safety Limits - N Reactor," dated August 3, 1972.
 3. Letter, C. D. Harrington to R. Ritchie, "Operating Safety Limits - N Reactor (DUN-8140)," dated March 23, 1973.
 4. UNI-15, "Operating Safety Limits - N Reactor," Staff of United Nuclear Industries, Inc., June 1, 1973.
 5. Letter, R. Ritchie to C. D. Harrington, "N Reactor Technical Specifications," dated November 13, 1972.
 6. Letter, O. J. Elgert to C. D. Harrington, same subject, dated May 31, 1973.
 7. Letter, N. R. Miller to O. J. Elgert, same subject, dated June 29, 1973.
 8. Letter, O. J. Elgert to N. R. Miller, same subject, dated October 11, 1973.

The major objectives of the technical specifications program that UNI has initiated are as follows:

- ① Respond in an acceptable manner to the AEC Manual Chapter 0540 (039k) requirements for the preparation of technical specifications without entering into a program of developing new technology.
- ② Prepare technical specifications that are as similar as possible to Regulation specifications in content, format, terminology, and language using the guides recommended and provided by AEC-RL and AEC-HQ.
- ③ Prepare a technical specifications document that will replace the Operating Safety Limits and which will be administered within UNI in the same manner as the Operating Safety Limits.
- ④ Construct specifications of sufficient latitude that no (or minimal) changes in the UNI Nuclear Safety Administration System will be required to adequately implement the technical specifications.
- ⑤ Improve the nuclear safety status of N Reactor operation by:
1) preparing technical specifications that reflect the knowledge of past operating experience and recent technological developments to the greatest extent possible; and 2) providing a considerable educational benefit to the UNI staff.
- ⑥ Show an "easily understood comparability" between health and safety administration for N Reactor operation and a typical commercial power reactor.

APPENDIX A

CHANGES IN PLANT SYSTEM DESIGN

Design Change 3070, Flow Monitor Tubing Replacement

This design change authorizes the replacement of the stainless steel sensing lines to the process tube flow monitors with Inconel tubing. All but one of the reactor right side, vertical tube runs were replaced during the 1970 Summer Outage. As the failure rate has not progressed as rapidly as anticipated the work is now being done on an "as required" basis. A visual inspection of the exposed portions of the sensing lines is made each scheduled outage.

It was decided that the sensing lines connected to the emergency cooling system (ECS) water flow monitors (four monitors on each side of the reactor) should be replaced with Inconel from the venturi connection to the flow monitor transducers. Because of the shortage of pipe fitter exposure, only a portion of this work has been completed. The vertical tubing runs are all in place, but not connected at the venturis. The horizontal tube bundles have been laid in the trays inside the thermal barrier as far as the outside wall of the thermal barrier. There has been no further progress this past year.

Design Change 3124 - Rod Scram Reset Circuit Modification

Several valves located in Zone I were rarely operated (V-5-1 and 2, V-6; V-13 and V-26). Equipment maintenance Standards (EMS) requires the operation of these valves to actuate the limit switches functioning in the rod scram reset circuit. Operation of several of these valves was done manually and in all cases, the operation of these valves required manual reseating.

Design Change 3124 authorized the removal, relocation and spare-off of those wires necessary to remove the valve relay contacts from the rod scram reset circuit, thus also eliminating the EMS requirement of periodic valve operation.

The work was completed and accepted for beneficial use in August, 1973. An addendum to the design change providing for the removal of additional valves from the rod scram reset circuit is now being prepared.

Design Change 3140 - Emergency Power Supply to ECS Valves, 100-N

Valves DWV-816 and FWV-817, provide for the flow of additional demineralized water and filtered water to the reactor during a condition requiring operation of the Emergency Cooling System (ECS). This permits the use of demineralized water and filtered water to the maximum extent possible before raw water is pumped through the reactor. Design Change 3140 authorizes the change of the power source for these valves from the A Bus to the emergency power supplied

by the 182-N diesel to insure a source of power to operate the valves during an emergency condition.

The work was completed in March of 1973.

Design Change 3168 - 100-N Ventilation Stack Monitoring Instrumentation

Design Change 3168 authorized the addition of a sampling system and the instrumentation required to routinely monitor the 100-N ventilation stack radionuclide effluents and to upgrade the existing air flow, radioiodine and particulate monitoring systems. This work was completed in 1971.

Addendum #1 to Design Change 3168 was prepared in 1972 and provided for the transmission of recording and alarm signals to the 105-N Control Room and for a redundant high level radiation monitor channel and a low level radiation monitor channel in the stack. This work was completed in March of 1973.

The complete system provides monitoring of air flow effluents and redundancy to insure that at least one channel will operate to record stack effluent radiation level, should an unusual incident occur.

Design Change P-3221 - V-12 Valve Replacement Spools - 105-N

The original Design Change 3221, issued June 26, 1972, authorized the installation of a total of 50 spools where diversion was not an operational requirement and where the C Hub (diversion header connection) was severely damaged. Thirty of the 50 spools authorized were installed.

Supplement 1 to Design Change P-3221 authorizes the permissive installation of V-12 valve replacement spools for leaking or inoperable V-12 valves, regardless of the C Hub condition. This decision was based on the conclusion that the retention of diversion capability for all replacement V-12 valves was not required.

Supplement #2 to Design Change P-3221 authorizes the permissive use of carbon steel forgings in addition to the welded replacement spool assemblies. Supplement #3 is now being prepared and it authorizes the fabrication of replacement V-12 spools with an accommodation for a Zone Temperature Monitor (ZTM) Resistance Temperature Detector (RTD).

To date, 92 replacement V-12 spools have been installed.

Design Change 3223 - B Bus Transient Power Protective Relaying

Design Change 3223 authorized relay modifications, which provide full protective features for parallel operation of B Bus with either D Tie Line or A Bus. This was required because during reactor down periods, D Tie Line is often paralleled with B Bus to provide a backup power source and during a transfer of power loads from A Bus to B Bus, the two buses are paralleled for short periods of time.

The directional overcurrent relays installed by Design Change were acquired from plant excess equipment and did not contain target coils to annunciate relay trip operation. Supplement #1 to Design Change 3223 authorized the purchase and installation of these target coils.

The bulk of the work has been completed and accepted. All will be completed early in 1974.

Design Change 3225 - Safety Circuit Simplification

The safety circuit trips from the intermediate and high level flux monitoring chambers had a time delay, using a special relay between the chamber amplifier and the safety circuit logic relay. Project DCE-539 provided for the installation of a new flux monitoring panel, which had modules with a "built-in" time delay so that the "built-in" time delay was cascaded upon the interposing relay's time delay. Design Change 3225 authorized the removal of the time delay relays from the intermediate and high level flux trips into the safety circuit. All work was completed in 1973.

Design Change 3226 - Combustion Engineering (CE) Boiler Control Power Isolation

Design Change 3226 authorized the installation of a separate transformer power feed to the CE boiler control systems. This improved the reliability of the CE boilers and minimized the recurring incidence of boiler flame-outs and trip-offs. The work was completed in April, 1973.

Design Change 3232 - Seismic Cubicle Supports, South Storage Basin - 105-N

Design Change 3232 authorized the addition of lateral supports in the south metal storage basin so that metal, contained in baskets, may be stored on top of the cubicles, when the cubicles are already filled with loaded canisters. The work was completed for beneficial use in May of 1973. Additional supports had been previously added to the north basin under Design Change 3208.

Design Change 3236 - Pony Motor and High Pressure Injection Pump Indicating Light Circuit Revision

A permissive switch was left in the off position and a high pressure injecting pump failed to start on demand. A need for positive identification where a

permissive switch is in the "off" position was apparent. Design Change 3236 authorized the installation of a new permissive switch with an additional contact in the green indicating lamp circuit. This contact will open the green indicating lamp circuit when the permissive switch is in the "off" position. The green lamp being off at all stations will indicate that the motor is not ready for operation. This installation was completed in 1973.

Design Change 3237 - 1310-N Radioactive Chemical Waste Handling Facility Piping Modifications

Routine surveillance revealed a loss of stored radioactive materials from the 1310-N Radioactive Chemical Waste Handling Facility. The leak was traced to the 10 inch buried recirculating return line. Design Change 3237 authorized the abandonment of this section of buried line and rerouted the return flow above ground from the exposed recirculation pump discharge to an unused 10 inch stub end installed for a potential second storage tank. This permitted the installation to be corrected with a minimum of expense and personnel exposure by avoiding the excavation, replacement and backfilling of the contaminated line. The work was completed early in 1973.

Design Change 3238 - Manual Trips for the 105-N and 109-N Confinement System

The 105-N and 109-N Confinement Systems possess redundant electrical circuits which automatically initiate Zone I confinement following pressure excursions. No ready single manual means of initiating confinement had been provided. Design Change 3238 authorized the installation of switches in the control room to trip each half of the redundant confinement electrical systems. This permits N Plant Operations to quickly manually trip the Confinement System in the event of an emergency and affords compliance with IEEE-279 which requires that Nuclear Plant Production Safety Systems have a manual means of actuation. A second benefit is the simplification of Confinement System testing. This work was accepted for beneficial use in August, 1973.

Design Change 3239 - Turbine Generator Condenser Coolant-Side Priming and Extraction System

Flow instability in the turbine generator cooling water system created operating instability in the turbine power generator operation upon loss of the "A" Bus. Testing revealed that a congregation of air was being collected in the Circulating Raw Water (CRW) System condenser discharge elbow, which disrupted cooling flows to the condenser. An operating mode, using in part, a large CRW flow obtained by reducing flow from the dump condensers, was devised. This mode did overcome the unstable condition but was not consistent with the AEC-RL request to establish the modifications necessary for reactor operation in the plutonium only mode. Design Change 3239 authorized the installation of a turbine generator condenser coolant-side air extraction system and other features necessary to provide a stable coolant flow at reduced CRW system head.

The equipment was installed, functionally tested and accepted for beneficial use in July, 1973 and the operational test was successfully completed and a final report issued (PT-NR-125, Suppl. 6) in November, 1973. (See also pages 14-17.)

Design Change 3240 - Roof Vacuum Breaker Maintenance Windows

Design Change 3240 authorized the installation of a maintenance window in one of the Confinement System Roof Vacuum Breakers. This was done as a prototype modification to assure that repairs could be made without removing the complete assembly. Supplement #1 authorized the installation of maintenance windows in the other three roof vacuum breakers. This work was completed in March, 1973.

Design Change 3249 - Ball Channel Funnel Liner Installation

Shrinkage of the 100-N Reactor ball channel liner blocks due to extended radiation has progressed to the point of nearly causing disengagement of the funnel and the top of the liner block in the central zone ball channel. Permissive Design Change 3249 authorizes the installation of funnel liners sized to fit into the liner block some 9-3/4 inches deeper than the funnel itself. This approach is not only more economical than replacing the whole funnel but also overcomes the difficulty of disposing of the highly radioactive replaced funnel. Twenty-nine funnel liners were installed during the 1973 summer outage. (See also page 9.)

Design Change 3250 - Replacement of Horizontal Rod Tip Inserts and Springs

Examination of two horizontal rod tip inserts revealed pitting corrosion of the aluminum extrusions and small amounts of boron carbide found in the dead legs of the rod coolant recirculation systems showed that the boron carbide is being washed out of these penetrations. Gradual loss of the boron carbide threatened to compromise the ability of the horizontal rods to consistently and predictably absorb neutrons. Permissive Design Change 3250 authorized the replacement of the horizontal rod aluminum tip sections. This work involved the successful offsite procurement of the special aluminum extrusions, the boron carbide and the actual assembly and filling of the tip sections. The onsite work included the rod removal, disassembly, reassembly and disposition of the removed radioactive tip section. The work was completed on the 83 rod positions covered in this design change. The other four rod positions are handled under differing circumstances. (See also page 23.)

Design Change 3252 - Pressurizer Pressure Control System

Design Change 3252 authorized modification of the Pressurizer Pressure Control System so that PRV-203-1 (Cycling Spray Valve) will be modulated from the pressurizer controller to control primary loop pressure rather than being cycled from the High-Low Alarm units. The PRV-203-1 valve was completely

overhauled and the continuous blowdown valve was adjusted to provide about 1cfm blowdown. The cycling vent valves RVT-203-1 and 2 are now operated normally closed. This has provided a more reliable and more stable pressure control during normal reactor operation. The work was completed and accepted for beneficial use in July, 1973. Further modifications are being made by Design Change 3286 to improve automatic control of pressure during the scram transient. The goal is to eliminate the need for operator adjustment during the scram transient.

Design Change 3254 - Replacement of Horizontal Rod Tip Sheaths

Permissive Design Change 3254 authorizes and documents the procurement of ten additional spare horizontal rod tip sheaths having a different grade of titanium and a possibility of greater ovality than originally specified. Successful procurement was completed in May, 1973 and three rod positions received new sheaths during the 1973 summer outage. (See also page 23.)

Design Change 3259 - Rupture Monitor Toggle Valve Reduction

An analysis of the Rupture Monitor System operational requirement revealed that two of the existing toggle valves on each channel could be eliminated without limiting the system's capabilities to detect ruptures. Design Change 3259 authorizes the removal of these two valves on each of the 1003 channels and the plugging of the valve openings. This change reduced significantly (40%) of the toggle valve repair program during the 1973 summer outage as well as reducing the potential of future leaks. The work was essentially completed during the 1973 summer outage.

Design Change 3261 - Modification of the CV-3, CV-5 and PCSV-202

Inspection of one CV-3 and one PCSV-202 revealed that the valve bushings and shafts were severely worn and the bushings were breaking into pieces. An emergency maintenance program was instituted to inspect and repair all of these valves during the remainder of the 1973 summer outage. Design Change 3261 authorized and documented the dimensional changes necessary for reworking component parts. Emergency procurement for the necessary components was initiated and successfully completed. Special tooling was developed successfully on site. All sixteen CV-3 valves and all six PCSV-202 valves were inspected, disassembled and reconditioned. Three of the CV-5 valves were inspected and were found to be in good condition. No other CV-5 valves were inspected. This work was completed during the 1973 summer outage. (See also page 23.)

Design Change 3273 - High Pressure Injection Pumps Automatic Stop Elimination

Design Change 3273 authorizes the removal of the automatic stop function for the third and fourth High Pressure Injection pumps. The automatic stop feature

(on a decrease in total flow) for the third and fourth HPI pumps has little utility for operation of the HPI system. The retention however, restricts the operator's freedom of override control and provides the possibility of failures which could prevent starting of the third and fourth pumps when needed. The DC was approved in December 1973 and is scheduled to be completed during the next outage.

Minor Design Change 72-34 - Flow Monitors for Crib Waste

MDC-72-34 authorizes the modifications necessary to record flow measurement through the twelve inch and thirty-six inch contaminated drain lines to the crib. Modification 1 to the MDC provided for the capability of removing the probes from the 12 inch and 36 inch lines for repair or replacement without draining the line. All work has been completed, tests run and installation accepted in 1973.

Minor Design Change 72-94 - Revised Safety Relief Protection for Recuperative and Spill Heat Exchangers

There are three safety relief valves protecting the recuperative and spill heat exchangers. In September, 1972 the inlet pipe to one of these valves failed and the reactor was shut down to repair the leak. An analysis revealed that the break was caused by fatigue and redesign of the protection system was indicated. MDC-72-94 authorized revision of the safety relief protection system to meet the minimum intent of the ASME code for unfired pressure vessels. The work was completed and accepted in 1973.

Minor Design Change 73-7 - Shield Cooling Sampling

The previous method of obtaining a thermal shield coolant sample required an operator to enter a high radiation (300 mr/hr.) zone in a hazardous location. MDC-73-7 provided for the sampling station to be moved to a less hazardous location where the exposure rate is much less (approx. 35 mr/hr.). The work was completed in February, 1973.

Minor Design Change 73-39 - Mini-Computer Installation

A Mini-Computer System for processing reactor operating data has been installed. MDC 73-39 authorizes relocation of components and panel revisions to permit the mini-computer installation in a convenient location in the 105-N control room.

Minor Design Change 73-42 - Rerouting of Steam Generator Blowdown and Drain Lines

In the past blowdown and drainage from the secondary side of the steam generator were discharged directly to the river through the 102 inch outfall line. Thus, it was possible to release small amounts of radioactivity depending upon system conditions. MDC-73-42 authorized an installation to provide the means so the blowdown and drainage from the steam generators, when contaminated, can be directed to the 1301 crib. The work was physically completed in 1973 but some ATP work remained to be done at years end.

Minor Design Change 73-45 - Traveling Wire Flux Monitor Effluent Isolation

Reactor tests have identified the outlet shield cooling effluent as a contaminated stream resulting from a cross flow or backflow cooling from the Traveling Wire Flux Monitor (TWFM) effluent stream. Minor Design Change 73-45 authorized the piping modifications necessary to isolate the TWFM effluent stream and route it to the 1301 crib. The work was completed and accepted in 1973.

Minor Design Change 73-55 - Effluent Sampling for the 102 Inch Outfall

MDC-73-55 authorizes the installation of a permanent sampling system on the 102 inch outfall. The original installation functioned properly but lowered river water levels reduced sample flows. Addendum #1 to MDC-73-55 provided for an additional collector in the next downstream vent of the 102 inch outfall. This collector has been installed but the ATP cannot be completed until the river level approximates the low water levels normally experienced in late summer.

Minor Design Change 73-72 - Seismic Anchors for the 181-N Gantry Crane

At the present time there is no positive device, other than rail clamps, to resist overturning the 181 gantry crane which could damage the river pumps during a severe earthquake. The rail clamps are inadequate to prevent tipping during a severe earthquake. MDC-73-72 authorizes fabrication of seismic anchors and the installation of these on the 181-N gantry crane. The MDC was approved in December, 1973 and is scheduled for completion early in 1974.

APPENDIX BUNUSUAL OCCURRENCESDrain Line Pipe Failure, January 29, 1973

During startup on January 29, an in cell inspection at 500 MW revealed a leak of undetermined origin on the 26" PCR line to cell 6 where the line passes through the sump between the pipe gallery and the cell. The reactor was subsequently shut down and an inspection showed that a 1" crack in the 2" drain line from the PCR line to drain valve PCRV 6204-9 was the source of leakage. The crack was a fatigue failure in the 2" line at the weld between the 2" line and PCR header which was caused by a stress condition that developed through thermal expansion and contraction of the components. The leak was repaired and a minor modification was made to eliminate the stress condition on the 2" drain line.

Ball Hopper Drops, January 29, 1973 and February 16, 1973

During reactor startup on January 29, 1973, a sudden reactivity decrease occurred, reducing reactor power level from 425 Mwt to 350 Mwt. It was determined that a ball hopper gate had opened, releasing poison balls to Ball Safety System Channel No. 38. The reactor was manually scrammed, the balls were removed from the channel and the channel was removed from service, after which startup was re-initiated. Subsequently, on February 16, when the reactor was operating at equilibrium power, Ball Hopper No. 61 spuriously actuated, producing primary cooling system transients which resulted in a reactor scram. Upon investigation of Ball Hopper No. 38, it was found that one of the two coils which holds the gate mechanism had failed; for Hopper No. 61, it was found that one of the holding coils had been de-energized. The release mechanisms for the ball hoppers are designed such that a single holding coil should be capable of retaining the gate in a closed position. Investigation and extensive testing revealed that in order for the holding coil to develop maximum force and hold the hopper gate closed it is necessary that the holding coil armature be fully seated in the holding coil. All hopper mechanisms were inspected and adjusted as necessary to ensure that the tripping mechanism would permit proper seating. In addition, procedures have been adopted for testing each holding coil following locking and cocking of the trip mechanisms prior to startup.

Annunciator System Out-Of-Service, February 10, 1973

During equilibrium N Reactor operation, smoke was detected coming from an annunciator power supply. Troubleshooting was immediately started to isolate the apparent short. Soon transformers supplying power to three other annunciator panels showed evidence of overheating. After a fuse in the power supply to one panel opened, the three other annunciator panels were removed from service. Under this condition, one of the normal means to monitor some plant conditions was not available, and substitute visual monitoring was placed in effect. No off-normal conditions were observed during the period of this special monitoring.

The short circuit was cleared by disconnecting the Recordalarm event recorder, which contains circuits common to all affected annunciator panels. The annunciator panels were returned to service, and circuits in the annunciator system were thoroughly checked for faults. The fault was located in an electrical plug on the Recordalarm and corrected.

No. 2 Exhaust Confinement Valve Malfunction During Valve Routine Testing
UO 73-6, March 2, 1973

After a normal reactor shutdown and during a routine confinement test, the No. 2 exhaust fan confinement valve failed to close automatically. Attempts to close the valve manually from the control room were unsuccessful but on a later attempt it functioned properly.

Laboratory testing and examination of the valve revealed the spring in the valve had a shorter free length than others in Spare Parts. All valves were checked to be sure the proper spring was installed as were the spare replacement valves.

Other backup measures were available had the failure occurred during an emergency confinement condition.

Ball Hopper No. 2 Binding During Testing, March 3, 1973

Scheduled tests were being performed on three N Reactor ball hoppers during the March outage. When the No. 2 hopper tripping mechanism solenoids were de-energized, the trip mechanism operated correctly, but the hopper gate was not lifted by the spring actuator and the balls were prevented from entering the reactor. After the trip mechanism was reset, the ball hopper operated in the normal manner. Tests of the other two ball hoppers were satisfactory. Following inspection of the stuck gate, the mode of failure was deemed to be wedging of the gate by the samarium oxide balls. Hopper test data indicate that this was an individual random failure and was the first gate mechanism opening failure since September 1970. The No. 2 ball hopper has been repeatedly tested satisfactorily and an additional nine ball hoppers were tripped and functioned properly. Based on these results, the ball safety system was considered to be in fully adequate operating condition. Total control and speed-of-control nuclear safety calculations allow for random failures of ball hoppers.

Five Rods Out of Reactor During Outage, March 5, 1973

During the March outage, a routine inspection of the N Reactor Control Room instruments by an oncoming shift revealed that five rods were out of the reactor, whereas the withdrawal of a maximum of three is permitted by the Process Standards when graphite temperatures are less than 100 F. Upon discovery of the out-of-standards condition, two rods were promptly inserted and appropriate management was notified. Subsequent analysis showed that with all five rods withdrawn, the reactor was 15 mk subcritical. Consequently, there was more inserted safety rod control available than the 10 mk shutdown margin required by the Operating Safety Limits. It was determined that the failure to follow Process Standards requirements resulted from a misunderstanding in communications between operating and maintenance personnel. Personnel have been reinstructed in the procedures to be followed when changing the status of plant critical systems.

Rupture Monitor Out of Service, March 12, 1973

During the initial phases of N Reactor startup from a scheduled outage, it was found that the rupture monitor systems were valved out of service, whereas the Process Standards require that they be in service before the reactor exceeds a power level of 5 Mwt. The condition was promptly corrected following discovery. Investigation revealed that the rupture monitor systems had been placed in service for reactor startup, but subsequently had been valved out of service for additional system maintenance. The approval for reactor startup had been based on the initial knowledge that the systems were in service. Personnel have been reinstructed regarding the use of startup check lists and the importance of prompt and complete log entries associated with changes in the status of equipment.

Broken One-Inch Line to the Radioactive Drain, April 2, 1973

Inspection of the 109-N pipe gallery during reactor shutdown revealed a broken one-inch drain line. The broken line was installed to drain an eight-inch spill cooler line valve and was connected to the radioactive drain (RDR). A valve in the line prevented release of primary coolant from the break; however, the break could release contaminated vapor/and or water from the RDR and from a 3/8-inch gland leak-off line to the Zone I atmosphere. No significant increases in activity were noted outside Zone I. The failure in the field run line resulted from fatigue, and since the line was not used, the line was removed and capped off.

Reduced Rupture Monitor Sample Flows for Two Process Tubes, April 9, 1973

N Reactor was shut down at the normal rate because rupture monitoring sample line flows for two process tubes were less than permitted by Process Standards. During the time that sample flows were reduced, indirect rupture detection capability at lower sensitivity level was provided by other gross gamma monitors. During the reactor outage, maintenance was performed on the system to return flows to normal.

Water Damage to Electrical Lighting Panel, May 3, 1973

On May 3, 1973, the spray from a broken water hose wet down lighting panel U via an open junction box, and the resulting short caused its feeder transformer to have shorted turns that tripped its main 480V feed circuit breaker. Power to panels UU and XX was lost because their 480V feed came from the same circuit breaker.

The loss of annunciator power in the right rod room and the loss of the segregation pit confinement door power prompted an investigation to review the circuits fed from these panels, and identify any critical circuits and relocate them to a more reliable power source. No critical circuits were found.

Loss of Power Supply to B Bus During Routine Switching Operations, May 19, 1973

During a reactor shutdown period, power is supplied from A Bus to B Bus through bus tie breakers. Power transfers were in progress to separate the two buses and allow the turbine generator (TG) or D Tie Line to supply B Bus. After the TG had been synchronized with B Bus and assumed a normal bus loading, the bus tie breaker was opened, separating the two buses. Within two minutes after the bus separation, the TG breaker opened, causing the loss of all power loads on B Bus. D Tie Line was switched on to B Bus and the loads were rapidly restored to B Bus. The loss of the TG was attributed to abnormal amounts of condensate in the turbine inlet supply. Corrections were made to steam traps to remove the source of condensate to the turbine inlet. Additional changes to the condensate system are planned during the 1974 summer outage.

Debris in Steam Generator 4A, May 23, 1973

The primary side of steam generator 4A was opened for inspection and to allow sampling of the tubing. Debris consisting of about three dozen pieces of metallic fragments were found lying on top of the divider plate and against the tube sheet. This debris varied in size from approximately 1/2" to 2-1/2" maximum dimension and included two 3/8" x 1-1/4" cap screws. Further investigation of all other steam generators, except 3A, led to the discovery of similar debris in each one. All debris found was removed from the steam generators and later identified as Stellite bushings from CV-3 and PCSV-202 valves and parts and pieces from the steam generator vent pipes. An inventory of the debris indicated that some pieces were lost in the primary loop; however, the unaccounted-for material does not present an unacceptable risk to nuclear safety or reactor operations since the nature and size of the missing debris is such that rapid blockage of any process tube is unlikely. A subsequent repair program returned the 16 CV-3 valves and the six PCSV-202 valves to an operable and reliable condition, and the steam generator vent pipes of all inspected steam generators have been removed since they are prone to failure and are unnecessary for proper steam generator operation.

Fire in Relay Panel PR-39, Reactor Power Setback Circuits, UO 73-17, June 12, 1973

During the 1973 scheduled summer maintenance outage, a fire occurred in relay panel PR-39 on June 12. The fire source was an overheated relay coil that ignited the plastic dust cover, which caused fire and smoke damage to 20 other relays in the same vertical row and the interior of the relay panel. The fire was detected and quickly extinguished by nearby technicians although the overhead smoke detector tripped and alerted the fire department. No critical nuclear control circuits were in the panel; so, all associated power was removed from wiring in the panel. An immediate review of the condition of plant systems led to the conclusion that the reactor was in a safe and controlled shutdown condition.

As a result of the accident investigation several recommendations were implemented for reducing and/or eliminating similar fire conditions. The major recommendation is to add a metal cover over the existing plastic cover. Prototype covers have been thoroughly tested and procurement of covers is in progress.

Momentary Loss of All A.C. Power to N Plant, July 4, 1973

The plant was in a shutdown mode with B Bus supplied from D Tie Line and A Bus supplied from the 50 MVA 230-13.8 kv transformer. A line to ground fault on the N Tap 230 kv line caused the automatic clearing of the 230 kv line section A4 to A9 by breakers opening at 100-D and 100-KE Substations. Five seconds after the fault clearing, the 100-KE Breaker automatically reclosed and restored power to A Bus.

When the fault on N tap line occurred, the D Tie Line Breaker (C1X1) tripped open. A first attempt by the operator to reclose C1X1 failed and a second attempt to reclose within one half hour was successful and power was then restored to B Bus.

All 230 kv relay and breaker operations were normal and without malfunction. The tripping of C1X1 Breaker was attributed to pilot wire relay operation during the 230 kv fault, although no relay target coil (annunciator) operated on C1X1. Pilot wire relaying is carried on telephone line circuits which run in the ground parallel to the 230 kv line from N Area. Ground fault currents induced into the telephone circuits were suspected as causing the pilot wire relay malfunction. The lack of relay target coil action was corrected by adjusting the target coil tap setting and retesting for proper action.

The fault on the 230 kv N Tap was caused by discharge of bird excrement over a 230 kv insulator support, creating a line-to-ground fault.

Bird guards have been installed in all sensitive areas to preclude birds resting in areas near the insulators. No problems have been encountered since installation of the guards.

Missing Spray Nozzles in Pressurizer, July 14, 1974

The N Reactor pressurizer was opened on July 14 for a routine internal inspection. The single-cycling spray nozzle and three of four backup spray nozzles were missing from the spray piping at the top of the pressurizer. Further inspection located one nozzle lodged on the heater bundle near the bottom of the pressurizer, and two nozzles lying on the bottom of the pressurizer next to the standpipe, which extends up into the pressurizer. The fourth nozzle could not be located, but it is almost certainly in the area behind the standpipe which cannot be viewed directly either with mirrors or the available TV equipment.

- *The standpipe is covered by a baffleplate such that there is no possibility that the nozzles could be carried into the reactor primary cooling system. Because the dislodged nozzles are extremely difficult to retrieve and because study indicated no nuclear hazard results from leaving them in the pressurizer, they were not retrieved. Five new nozzles were installed.

Diesel Engine Damage During Startup of No. 2 High Lift Pump, July 25, 1973

On July 25, 1973 N Reactor was in the 1973 summer outage. Rod calibration work was in progress, which required A Bus and B Bus in service and the emergency cooling system available for service. A ground occurred in a DC system control panel, and in accordance with Process Standards requirements, one of the three ECS diesel-driven high lift pumps was placed in operation to circumvent the possibility that the dc ground would disable the diesel engine starting circuit and prevent the engine from starting upon an emergency demand for reactor backup cooling water. Later the ground was traced to circuits not associated with the high lift pump circuits, and diesel engine operation was discontinued. At this time, it was discovered that water was leaking from one cylinder head. Inspection revealed damage to the piston, head, and cylinder sleeve.

Subsequently, the engine was repaired and returned to service. It was concluded that the damage had been caused by starting the engine with excessive coolant liquid in the cylinder (this type of damage is not unusual for this service). Procedures for detecting coolant are being implemented.

Momentary Loss of All A.C. Power to N Plant, July 31, 1973

The plant was in a shutdown mode with A Bus supplying power to B Bus. The T-G was not operating and D Tie Line Breaker was open. A momentary line-to-ground fault on a 230 kv tower near 100-D Substation caused the tripping of 230 kv Breaker at 100-KE and 100-D with total loss of power to N Area. The breaker at 100-KE reclosed in five seconds automatically restoring power to both A and B Buses at N Area.

The fault on the 230 kv line structure was attributed to discharge of bird excrement over a 230 kv insulator support. This fault was similar to the fault which occurred on July 4 (see page 47) except that it occurred on a different location of the line (A4 to A9). All 230 kv relay and breaker operations were normal and without malfunction.

The N Reactor was in an extended scheduled outage. Normal recirculation cooling was being provided by primary pump pony motors. During the 4.5 seconds of power interruption, the pumps were without power. All returned to proper operation when power was restored. The emergency cooling system was available for backup cooling. The power and reactor cooling system arrangements were normal and in accordance with Process Standards.

Reactor Scram on Unbypassing Flow Monitor Module, August 9, 1973

N Reactor had been started up on August 9, and on August 10 the power level was being held at 3300 Mw to analyze and correct a flow monitor system problem. Trip signals were being received on the system, which indicated that one or more of the individual flow monitor modules was near its trip point. Indicated flow to each individual process tube was measured with an automatic scanner and determined to be greater than eight percent from the trip value. One module suspected to be causing the trouble was bypassed, while coolant flow and temperature were monitored on other instrumentation. The module trip circuits were tested, and it appeared to be normal. However, after the original trip settings were restored and the module was unbypassed, a flow trip indication occurred, causing a reactor scram even though flow to the tube was normal.

The module causing the trip and another suspect module were removed and replaced with reconditioned modules.

Loss of Graphite Cooling Surge Tank Level Control, August 11, 1973

On August 11, the N Reactor graphite cooling system (GCS) surge tank level control became erratic, and the reactor was shut down. Corrective measures were taken that appeared to correct the problem. The reactor was restarted, and the GCS surge tank level control appeared normal. On August 12, it was found that the surge tank was full (8.5 ft.) despite indications of low water on the indicator. At about the same time, a pressure variation caused a GCS relief valve to open and release water at a rate of approximately 10 gpm. A feed-and-bleed control method was established to regulate system pressure and allow the relief valve to seat. Subsequent analysis has shown that level control problems result from introduction of air into the system during maintenance and testing. Additional gas makeup and venting capacity has been added and other corrective actions are planned.

Breaks in Deactivated 1 1/4-Inch Low Pressure Drain Lines, August 30, 1973, September 10, 1973

An inspection of the right outlet pipe space revealed a broken 1 1/4-inch low pressure drain line between a V-58 valve and the 16-inch low pressure flush line. In a subsequent inspection during an outage on September 9, a nearly identical break was found in an adjacent line from another V-58 valve. The breaks occurred at the welded couplings on the flush line and subsequent analysis indicated they were caused by fatigue. The breaks resulted in release of low level contaminated coolant to Zone 1 which caused higher than normal humidity.

The failed piping was marginally supported and this condition apparently was conducive to excessive vibration and, therefore, fatigue failure. A program has been initiated to identify and correct deficiencies in installation of all drain and vent valves. It should be noted that this and other similar pipe breaks reported previously have all involved 2-inch and smaller piping. This pipe size was "field run" during plant construction and hence, the details of supports were not necessarily well engineered. Therefore, even though a 2-inch hole in the primary coolant system would not affect continuity of core coolant, program efforts will be focused initially on that system. The balance of the plant systems will be inspected as outage time and personnel exposure permit.

Control Malfunction of Raw Water Supply to the Spill Cooler, September 7, 1973

During equilibrium operation of the reactor on September 7, a series of events led to the conclusion that the supply flow of raw water to the spill cooler was excessively high due to malfunction of the automatic temperature control system. Consequently, the system was placed on manual control.

During the outage starting on September 9, further troubleshooting of the system was enabled by accessibility to Zone 1 and two conditions bearing on the malfunction were noted:

- ① The control signal air supply line to one of the two automatic control valves had pulled out of the fitting to the valve, causing it to be in a closed position.
- ② A packing leak on a pressure relief valve on the raw water supply caused water to spray into open conduit junction boxes containing the spill cooler control RTD leads. One of the two redundant RTD assemblies was wet but both gave erratic temperature indications.

Both of the above conditions could have affected the operability of the control system and were corrected prior to the subsequent startup. However, spill cooler temperature control has remained on manual operation to assist in controlling water hammer in associated piping. Corrective actions are planned.

RV-2 Valve Out of Service, September 10, 1973

During an unscheduled outage on September 10, it was discovered that a drain valve, V-57, on the pilot valve for RV-2-1 primary loop relief valve was closed. The closed drain valve rendered the RV-2-1 valve inoperable. Primary loop relief valve RV-2-2 was in service and had adequate capacity to provide system pressure relief should that have been required. Procedures for locking RV-2 valve ancillary valves have been implemented.

Gas System Security Valve Closure, September 21, 1973

During reactor operation on September 21, it was discovered that the reactor gas atmosphere activity had been increasing for approximately two and one-half hours. The increase in gas activity violated provisions of PCA-N-49-73 which permitted temporary substitution of the gas activity instrumentation for the gas chromatograph. A failed security valve between the high pressure and low pressure helium systems resulted in a lapse of gas makeup and thus gas pressure loss to the reactor which, in turn, permitted temporary air in-leakage. The small amount of air in-leakage would not be expected to significantly affect expected reactor moderator life. The security valve has been repaired and plans are being made to install low pressure annunciation on the low pressure helium supply system.

Spent Decontamination Solution Spill at 1314-N Loading Station, September 24, 1973

During the loading of a railroad waste tank car with spent decontamination solution at the 1314-N loading station, the waste solution overflowed from the tank car pipe fittings. The loading pump was turned off, following which the solution flowed up through the drain in the catch basin beneath the car into an adjacent drywell. Approximately 100 gallons of the solution flowed out of the drywell onto a 20' x 20' area of ground inside the radiation zone. The 100 gallons contained a total of 0.015 curie of radioactive contaminants. The cause of the incident was found to be failure to open a vent valve on the top of the tank car.

The contaminated soil was removed, packaged, and disposed of at a burial ground. The Operating Procedure was revised to include a checklist to assure that all steps in the loading process are completed. The design of the tank car loading equipment is being reviewed.

Boiler Steam System Malfunctions, October 5, 1973

Shortly after a scheduled reactor shutdown, when the reactor main steam header pressure was reduced, an abnormally rapid steam load demand caused flameout of the two in-service CE boilers. This in turn caused loss of steam to the turbine generator set and the primary pump drive turbines. Just prior to loss of the turbine generator set, B Bus supply was transferred to the D-Tie line. When the primary pump drive turbines shut down, the primary pumps were driven by pony motors supplied by A Bus, thereby maintaining adequate shutdown cooling by the primary cooling system. B Bus steam supply was returned to service in about one-half hour.

The increased steam load was caused by improper automatic closure of one or more nonreturn valves which are intended to prevent flow of steam from the turbine generator header to reactor main steam header when the main steam header pressure is reduced during reactor shutdown. Mandatory procedures which will not rely on automatic seal off of the nonreturn valves have been added to prevent excessive demands on boilers during main steam header depressurization. The nonreturn valves will be repaired during the next summer outage when the main steam header can be taken out of service.

Low Retention Efficiency, C Cell Particulate Filters, October 8, 1973

During the scheduled October 1973 outage, a routine inplace test of the 117-N C-Cell particulate filters was being conducted, as required by Equipment Maintenance Standard on a six-month interval. During the initial phase of the test (calibration of the photometers), the engineer operating the photometers reported an unusually high reading, about two percent, for the concentration of test smoke (di-octyl phthalate) downstream of the filter. At the same time, the engineer operating the sample probe at the downstream location reported visible smoke at the C-Cell outlet gate. Permissible penetration for in-service filters is <0.05 percent.

The C-Cell particulate filters were replaced with new units testing better than 99.99 percent efficiency. The new units will continue to be tested at six-month intervals as required by the EMS. One of the failed units is undergoing examination in an effort to determine the cause of failure. There are no reportable results from the examination at this time.

Circuit (CE-20) Energized During Reactor Operation, October 24, 1973

During a routine audit of the status of the 125 v DC circuits in Room 6 of the 105-N Building, it was noted the circuit CE-20 was energized from the battery cart auxiliary power supply. Power from this source and C battery circumvents the breaker for Circuit CE-20 which was turned off. Circuit CE-20 supplies power to unlatching circuits of Zone 1 confinement rear doors in the 105 Building. Grounds in these circuits can cause inadvertent unlatching of doors in Zone 1, and for this reason the Process Standard requires that these door circuits be de-energized during reactor operation.

Inspection of rear confinement doors showed that none of the doors had become unlatched up to the time the reported condition was identified and corrected. Thus, the condition did not represent an immediate nuclear safety hazard. Corrective actions include revision of startup checklist to check the status of the battery cart auxiliary power supply.

Secondary System Upset, November 20, 1973

During equilibrium reactor operation, Dump Condenser 14 was isolated for valve calibration work and other checking. When the dump condenser was returned to service, the secondary system surge tank dropped below normal level and the main steam header/condensate pump suction header ΔP oscillated from -5 to +9 psi which exceeded the Process Standard limit of ± 3 psi. During the process of returning the system to normal, two subsequent oscillations occurred.

In subsequent investigations it was determined that incorrect calibration of isolation valves resulted in improper preheating of the dump condenser and an abrupt opening of the valve to the condensate header, thereby causing a sudden flow of condensate from the surge tank and other dump condensers into the off-line condenser. Corrective actions include recalibration of the valves, and publicizing the occurrence and the potential for such a system transient.

Loss of Control Rod Coolant Recirculation Pump No. 2, November 15, 1973

During equilibrium operation, one N Reactor control rod coolant recirculation pump bearing was noted to be overheating, and an attempt was made to provide auxiliary cooling. Before the pump could be removed from operation, the bearing seized on the shaft and the pump failed. The one remaining pump continued to provide coolant to the right side control rods. The system is also designed for a nuclear safe transfer to once-through backup filter water coolant. Following repair of the pump when the pump was being returned to service, a flow deviation caused an automatic transfer to the backup coolant system. It was subsequently determined that the pump failure resulted from misalignment of the pump. A check of rod coolant recirculation pumps on the left side of the reactor revealed similar operating problems. When one of the pumps was removed from service, this system also transferred to once-through backup coolant. All pumps have been checked and restored to service, flexible joints have been installed at the inlet to the pumps, and the system returned to recirculation operation.

Delayed Detection of Fuel Failure, December 7, 1973

N Reactor was shut down at 0207 on December 7 because of general fuel failure indications. At the time of shutdown, the location of the failure had not been identified, but radiation levels at primary piping after shutdown confirmed that a failure had occurred. Radiation surveys and review of operating data indicated that process tubes 1860 and 2350 were the primary

candidates, although 26 other tubes were considered suspects. Because of the higher radiation level at the rear cap, tube 1860 was discharged first. This fuel was examined and no fuel failure was found. Tube 2350 was then discharged, and the failure was found to be the outer element of the fifth fuel assembly from the downstream end. The failed element was weighed later, and it was determined that approximately 1.6 pounds of uranium were missing. The 0.95 percent U-235 Mark IV fuel had been charged into tube 2350 during the 1973 summer outage and had attained 41 percent (100% Mwd/T) of goal exposure.

Investigation of the rupture monitor system after shutdown revealed that the valve in the sample line for tube 2350 had been left closed following a back-flush operation on its sample chamber following an indication of high radiation. Post-shutdown surveys revealed that radiation levels around the primary piping were from three to ten times normal.

Calculations indicate that approximately 220 Ci of I-131 were contained in the 1.6 pounds of uranium released to the primary coolant stream. Preliminary results of radioanalyses of samples of effluent water entering the 1301-N crib indicate I-131 concentrations on the order of 10^7 pCi/l, which is about 100 times the average concentration.

Preliminary results of Columbia River sampling performed by Battelle-Northwest Environmental Evaluations group indicate that the I-131 concentration at the Richland water plant intake did not at any time exceed one percent of the yearly maximum permissible concentration (MPC) for uncontrolled areas. The dose to Richland residents from this release will, therefore, be insignificant.

Corrective actions completed and in progress include:

- Control procedures for backflushing rupture monitor sample chambers have been put in effect, requiring that an operating supervisor instruct the instrument maintenance technician in writing the sample chambers to be backflushed. The technician then performs the backflushing operation, returns the valves to the normal position, and so informs the operating supervisor in writing. A second, independent check is made to confirm that the affected lines are properly valved.
- Operations personnel have been instructed to exercise the greatest degree of discretion in analyzing indications of possible fuel failures in order to minimize the possibility of continuing reactor operation with a failed element.
- Meetings were held with all operating and maintenance personnel having responsibilities for operation and maintenance of the rupture monitor system to review this unusual occurrence and to assure that all persons are informed of the new procedures described above.
- Additional backup fuel failure monitoring instrumentation has been installed, and revised definitions of bases for fuel failure confirmation utilizing the additional monitoring system have been developed.

- 0 Evaluation of the release of radioactive materials as a result of this incident will be completed and reported.
- 0 Procedures for minimizing release of radioactive material to the environment in future similar events have been developed and implemented.

Unidentified Fuel Element Reject at End Tester, December 10, 1973

Following the N Reactor fuel element failure on December 7, 1973, a search was made of fabrication and testing records for the failed fuel element. The cap-to-core bond tester strip chart recorder records showed that the fuel element had been identified as having a defective braze at one end of the fuel element, and the chart was marked accordingly; however, the test equipment operator failed to identify the fuel element such that the fuel element could be removed from the process stream following subsequent tests.

The test equipment operator, now retired, was considered an extremely careful and conscientious operator, and this lapse is completely unexplained. Corrective actions include: 1) a search of records to determine whether any similar fuel elements existed in the cold inventory or in the reactor; 2) institution of a double-check of all nondestructive tester charts by an independent group; and 3) publicizing the event and emphasizing the personnel responsibility for accurate and complete records of fuel fabrications.

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