PROD. & UTIL. FAC. 50-329,330

UNITED STATES OF AMERICA

ATOMIC ENERGY COMMISSION

BEFORE THE ATOMIC SAFETY & LICENSING BOARD

In the Matter of	Docket Nos. 50-329
	50-330
CONSUMERS POWER COMPANY	
	SWORN TESTIMONY OF
Midland Plant, Units 1 & 2	RICHARD E. WEBB

STATE OF OHIO)
)ss
COUNTY OF FRANKLIN)

Address:
1612 Andover Road
Columbus, Ohio 43212

I, Richard E. Webb, being first duly sworn, depose and say as follows:

CURRICULUM VITAE

Background:

1. B.S. Engrg. Physics, University of Toledo, 1962.

2. Division of Naval Reactors, AEC, 1963-1967, where my primary responsibility was for the nuclear reactor portion of the Shippingport Pressurized Water Reactor Plant.

3. Consumers Power Company, Big Rock Point Nuclear Power Station, associate engineer, worked mainly on reactor engineering duties. June 1967-December 1967.

4. Ph.D. candidate in Nuclear Engineering at the Ohio State University. Status--finished research and course work and finalizing the thesis. Subject: Autocatalytic Effects During Explosive Power Transients in Liquid Metal, Fast Breeder Reactors. Expect to complete the degree requirements in October, 1971.

5. Certificate of Successful Completion, Bettis Reactor Engineering School, 1965 Bettis Atomic Power Laboratory, West Mifflin, Pennsylvania (operated by Westinghouse Electric Corporation for the Atomic Energy Commission).

Author of:

1. "The Unconstitutionality of the 1954 Atomic Energy Act and subsequent amendments" self published and distributed.

2. "Treaty making and the President's obligation to seek the Advice and Consent of the Senate," Ohio State Law Journal, Vol. 31, No. 3 (1970). See Cong. Record, July 29, 1971 (S12059-12075).

OPERATING EXPERIENCES OF STEAM GENERATORS AND HEAT EXCHANGERS IN NUCLEAR POWER PLANTS AND THE IMPLICATIONS FOR THE MIDLAND [PRESSURIZED WATER REACTOR] PLANT, UNITS 1 AND 2

According to the Advisory Committee on Reactor Safeguards (Ref. 1) the Midland Plant is the first dual purpose reactor plant to be licensed for construction. That is, part of the steam produced by the steam generators within the reactor plant will be exported to Dow Chemical Company for use in manufacturing chemical products. The remainder of the steam will be used to generate electricity at the reactor plant. Because the reactor coolant will be radioactive, questions are naturally raised as to the possibility of (1) radioactive contamination of the Dow products and (2) radioactivity released to the environment, since 25% to 40% of the process steam will not be recovered. To prevent radioactive contamination of the process steam, it is necessary to isolate the process steam from the reactor coolant by at least one or a series of heat exchangers which must not develop primary to secondary leaks; i.e. the integrity of the physical barriers between the primary coolant and process steam must be maintained. This paper cites evidence of heat exchanger malfunction which should be useful in estimating what might be expected in the way of radioactivity leakage to the process steam (Items #1, 2, and 3 which are attached). Of course, the extent of fuel element cladding failure in the reactor will affect greatly any estimates of radioactivity leakage. However, this paper does not consider the question of fuel element performance capability for the Midland reactors, except as to recommend that an independent review be conducted as to the adequacy of the fuel element design for the planned power operations.

SHIPPINGPORT STEAM GENERATORS

The Shippingport Pressurized Water Reactor Plant operated with Core 1 (230 MWT) for about six years using four steam generators--two generators (1 A and 1 D) were of the straight tube design (Foster Wheeler Corp.) and the other two (1 B and 1 C) were of the U-tube design (Babcock and Wilcox Co.), (see Ref. 2) the tubing was made of stainless steel.

Item #1 briefly summarizes the experience with the Core 1 steam generators. It reports that leakage and widespread and extensive cracking occurred in one of the Babcock and Wilcox generators. Presumably, Bettis Atomic Power Laboratory (Shippingport designers) had submitted reports to the AEC concerning the performance of the Core 1 steam generators and the radiological consequences of the leaks. It is recommended that these reports be obtained. However, since Core 1 operated with no fuel failures (except for 3 tiny pin hole leaks in the cladding of the natural uranium fuel rods which were barely detectable), the extent of radioactivity leakage into the secondary system was minimized. (Ref. 3).

Shippingport Core 2 (505 MWT design power, but normally operated at 67% of full power) began operation in 1965. New steam generators were installed for Core 2 with greater heat transfer capacity to handle the increase in core power. The tubing was made of Ni-C_r-Fe alloy (Inconel) "to enhance the integrity of the system." (Ref. 4). (In Item #2, Byerley states that Iconel has been selected as a standard for present day steam generators over stainless steel for better corrosion resistance. Byerley also states that primary to secondary leaks occurred with the Rowe-Yankee steam generators.)
As with Core 1 the 1A and 1D units for Core 2 are straight tube, Foster Wheeler generators and the 1B and 1C units are U-tube Babcock and Wilcox generators.

Despite the switch to Ni-C_r-Fe alloy for tubing material, the Core 2 steam generators have experienced leaks. Item #3 is a chronological compilation of excerpts from the Shippingport Quarterly Progress Reports dealing with these leaks and related tritium activity levels in the primary coolant. The excerpts begin with WAPD-MRP-118 and end with WAPD-MRP-135 and cover the period from June 1966 to January 1971. The reports were issued by the Bettis Atomic Power Laboratory.

The highlights from these excerpts are as follows:

1. Both of the Babcocks and Wilcox' U-tube generators (1B and 1C) developed leaks and one of the Foster-Wheeler, straight tube generators (1D) developed leaks. The leak rates were as follows:

   1B: .21 gal/hr--25 gal/hr (129 ml/min--1570 ml/min) before the leak rate increased to 100 gpm.

   1C: 8-9 gal/hr

   1D: 1.6 gal/hr

   After the 1B generator was returned to service (after the leaking tubes were plugged) it again experienced a rapid leak of 150 gal/min and was then removed again from service for extensive investigation.

2. The cause of the leaks were not identified.

3. Bettis Laboratory submitted a comprehensive report to the AEC regarding the effect of steam generator primary to secondary leakage on reactor plant operation. (See WAPD-MRP-135, p. 1) Bettis described this report as follows:

   > The Bettis Laboratory submitted to the AEC for information during this report period an evaluation of the effect that various steam generator leak rates would have on continued plant operations. The effect of continued operation with a leaking steam generator was evaluated with respect to radioactive waste disposal system processing capabilities, contamination of the secondary plant, and eventual discharge of activity to the environment from both normal and postulated accident conditions. This evaluation provides

guidance as to when a steam generator with primary to secondary leakage should be removed from service for maintenance.

4. The tritium level in the reactor primary coolant reach a high of 55 micro-curies per liter ($\mu c/l$) and then decreased due to coolant leakage. The production rate of tritium was estimated at 2.4×10^3 $\mu c/hr$.

Based on these highlights the following comments are offered:

a. Although, not reported in the WAPD-MRP excerpts, one of the rapid leakages in the 1B generator almost resulted in the initiation of the Core Safety Injection System, which is otherwise known as the Emergency Core Cooling System. (Ref. 3) The loss of primary coolant caused a drop in primary pressure from 2000 psig to near 1300 psig, which is the trip point for Safety Injection. The question arises as to what would happen if the emergency cooling system were initiated by a rapid leak in the steam generator tubing. Would the result be a flushing of the primary coolant radioactivity into the secondary system? Are primary coolant isolation valves provided for each steam generator?

b. A tritium level of 55 $\mu c/l$ implies that approximately 4.7 curies of tritium total, were present in the Shippingport primary coolant:

$3000\ ft^3$ of coolant $\times (2.54\ cm/in)^3 \times (12)^3\ in^3/ft^3 \times 10^3\ l/cm^3 \times 55\ \mu c/l = 4.7$ curies/H^3

However, Core 2 operation has been marked by no detected fuel failures (see WAPD-MRP reports in general) and so it must be added that the radioactivity levels in the primary coolant system might, therefore, be considered minimum.

c. The report mentioned in WAPD-MRP-135, p. 1 would appear to be highly relevant to the concern for radioactivity leaking into the Dow process steam. It is therefore recommended that the Atomic Safety and Licensing Board obtain the report for review by all interested parties, including the Mapleton Intervenors. Mrs. Mayfield of the AEC's Division of Technical Information Extension at Oak Ridge (phone: 615-483-8611, ext. 34765) is attempting to locate the report within the AEC.

Big Rock Point Feedwater Heaters

The Big Rock Point Nuclear Power Plant (Boiling Water Reactor) had experienced considerable difficulty with feedwater heaters. One or more heat exchangers had to be removed from service because of a general rotting away of the tubes. These tubes were not made of either stainless steel or Inconel. It was planned to replace these heaters with new heat exchangers equipped with stainless steel tubing. (Ref. 5) It would be of interest to know whether th se new heat exchangers were placed in service and whether any l aks had developed within them.

References:

(1) Letter to Honorable Glenn T. Seaborg from the Advisory Committee on Reactor Safeguards, June 18, 1970 Subject: Midland Plant Report

(2) The Sh ppingport Pressurized Water Reactor, Addison-Wesl Publ (1958), by Personnel of the Naval Reactors Bra h of the AEC; Westinghouse, Bettis; and Duquesne Ligh Co., p. 33

(3) Personal recollec ion based on my association with the Shippingport program.

(4) PWR Core 2 Reactor [Shippingport] Design Description Report, R. Atherton, et al, WAPD-296, March, 1968, p. 134. See also, PWR Core 2 Safety Analysis, WAPD-Sc-501 (Del.) June 1964.

(5) Personal recollection based on my association with Big Rock Point.

Further deponent saith not.

RICHARD E. WEBB

Subscribed and sworn to before

me this 21 day of September, 1971.

[illegible]

ITem #-1

959 CORROSION OF STEELS AND IRON-CHROMIUM-NICKEL ALLOYS BY LIQUID SODIUM. Champeix, L.; Carmec, [illegible] (CEN, Saclay, France). Bull. Inform. Sci. Tech. [illegible] No. 135, Sept.-June-Aug. 1969. (In French).

[illegible] a few general remarks on the mechanism of corrosion by liquid metals, which are noticeably different from those in the more conventional media, the problems connected with the corrosion of various categories of material considered are examined [illegible]. Austenitic steels are commonly considered to be reasonably resistant to liquid sodium so long as the oxygen content of the latter is kept reasonably low. However there is some controversy over the fundamental corrosion process, and the kinetics of the phenomenon are thought to differ. Ferritic steels behave similarly if they contain enough chromium to lessen the tendency toward decarburization. (auth)

960 FINAL EVALUATION OF SHIPPINGPORT POWER STATION'S STAINLESS STEEL GENERATOR AFTER FIVE YEARS. Wel[illegible], I. H. (Westinghouse Electric Corp., West Mifflin, Pa.); Brindley, W. F.; Berry, W. E.; Ward, S. F. pp 1-13 of High Purity Water Corrosion of Metals. Houston, Tex., National Association of Corrosion Engineers, 1968.

From 23rd Annual Meeting of the National Association of Corrosion Engineers, Los Angeles, Calif. See CONF-670312.

The original four steam generators used for more than five years in the Shippingport plant were tubed with Type 304 stainless steel. Leaks developed in the 1B unit after only a short period of operation. Results of an investigation at that time lead to the conclusion that a combination of free caustic in the boiler water and a prevailing concentration mechanism caused the transgranular cracking observed in removed sections of tubes. The same 1B unit, removed from the plant after over five years' service, has now been subjected to a complete destructive examination. The condition of the unit and the boiler water control used is described. Widespread cracking was found in the tubes. The cracks were intergranular in the sensitized sections and transgranular in the annealed sections. Although the cracking could be attributed to either chloride or caustic stress corrosion, the evidence points to caustic as the more probable cause. Most of the cracking apparently occurred during the initial 150 hours of operation when the boiler water contained free caustic. Seventeen of 31 [illegible] tubes from the 1B unit, when examined in detail, showed [illegible]. [illegible] and monosodium phosphate were used to [illegible] rather than caustic soda; disodium phosphate was substituted for [illegible] pyrophosphate and phosphate concentration was decreased. (auth)

961 INDUCTION OF CREVICE CORROSION IN REACTOR MATERIALS BY CONCENTRATION OF COOLANT ADDITIVES. Lehmann, W. (Westinghouse Electric Corp., West Mifflin, Pa.); Ernst, P. E.; [illegible], G. G. pp 14-21 of High Purity Water Corrosion of Metals. Houston, Tex., National Association of Corrosion Engineers, 1968.

From 23rd Annual Meeting of the National Association of Corrosion Engineers, Los Angeles, Calif. See CONF-670312.

Tests performed with internally heated crevice corrosion test specimens, simulating possible Shippingport (PWR) reactor crevice geometries, show that non-volatile coolant additives could concentrate in boiling crevices, inducing accelerated corrosion. Chemical analysis of solutions instantaneously trapped in a boiling crevice show concentration factors of up to 1000 over a ten-day period. Rate of concentration is a function of both temperature and geometry. Results obtained from Zircaloy crevice corrosion testing performed at temperatures above T_{sat} are presented for different crevice geometries. A theory for the mechanism of concentration has been substantiated in part by experimental data. (auth)

962 DISPERSION OF CORROSION PRODUCTS FROM A FILM-FORMING METAL EXPOSED TO AN AQUEOUS STREAM. Durham, J. W. (Moffett Naval Air Station, Sunnyvale, Calif.); [illegible], T. J.; Mills, C. M. pp 22-34 of High Purity Water Corrosion of Metals. Houston, Tex., National Association of Corrosion Engineers, 1968.

From 23rd Annual Meeting of the National Association of Corrosion Engineers, Los Angeles, Calif. See CONF-670312.

Transport and dispersion characteristics of corrosion products from cold-rolled, mild steel exposed to flowing high purity water were investigated theoretically and experimentally. Flat-plate specimens with nonreactive leading portions were exposed in a carefully controlled hydraulic regime to aerated, distilled water at a temperature of 140°F and Reynolds numbers in the range 1000 to [illegible]. Corrosion product accumulation as a function of position along the specimen was determined using ^{59}Fe as a tracer. By comparing a digital computer analysis of the rigorous equations and a simplified model to the data, results are shown to correspond to the transport equations that describe a process in which soluble species coagulate to form particulates. The data and analysis indicate that corrosion product mobilities are velocity dependent but very low in the range investigated. (auth)

963 ON THE INTERNAL AND EXTERNAL OXIDATION OF Cu-1 PERCENT Zr, Cu-0.5 PERCENT Ti, AND Cu-1 PERCENT Ti ALLOYS. Vasu, K. I. (Indian Inst. of Science, Bangalore). Corr. Sci. (India), 9: 477-490(Oct. 20, 1969).

Oxidation of Cu-1% Zr, Cu-0.5% Ti and Cu-1% Ti alloys in pure oxygen in the range of 400 to 900°C was generally governed by a linear rate law in the initial stage (which is very conspicuous with Cu-1% Zr alloy) and a parabolic rate law in the succeeding stage. The linear oxidation had a low activation energy of 5 kcal/mole and involves a phase-boundary process of chemisorption. The parabolic oxidation, controlled by the Cu^{+} ion diffusion through the scaling layer, has a high activation energy of 50 kcal/mole. (auth)

964 A METHOD FOR THE PROTECTION OF ZIRCONIUM AND ITS ALLOYS AGAINST THE ABSORPTION OF HYDROGEN. Schleicher, Hans-Walter; Brossa, Francesco (to EURATOM). British Patent 1,165,05[illegible]. Oct. 29, 1969. Priority date Aug. 12, 1966, Belgium.

Methods of protecting Zr and its alloys from hydrogen embrittlement are described. The metal surface to be protected is covered using aluminum-silicon solid solutions and heated at 450 to 650°C. A thin film interlayer of metal such as copper may be used. (J.R.D.)

965 THE EFFECT OF HEAT TRANSFER ON THE CORROSION BEHAVIOR OF TYPE 304 STAINLESS STEEL IN BOILING WATER. Steigerwald, R. F. (Climax Molybdenum Co., Pittsburgh). Contract AT(07-2)-1. Trans. Met. Soc. AIME, 245: 2175-8(Oct. 1969).

The effects of heat transfer on the corrosion behavior of Type 304 stainless steel in boiling water were studied. Heat transfer conditions increased the tendencies of the stainless steel toward stress-corrosion cracking when the water was contaminated with Cl^{-}. Surface preparation was the most important variable in determining the severity of the stress-corrosion problem in water with a given Cl^{-} content. Heat treatment, chemical cleaning, and degree of wet film boiling also affected the corrosion of stainless steels used as heat transfer surfaces in boiling water. An apparatus for corrosion testing under controlled heat transfer conditions is described. (auth)

Preparation and Fabrication

Refer also to abstracts 351, 995, 1029, and 1072.

966 (GEMP-1012(Pt.2), pp 13-30) PHYSICAL METALLURGY OF FAST BREEDER REACTOR CLADDING MATERIALS AND REFRACTORY METALS. Collins, C. G.; Schlanger, K. M. (General Electric Co., Cincinnati, Ohio. Nuclear Systems Programs).

Studies to define thermal-mechanical treatments yielding optimum ductility in LMFBR fuel-cladding alloys were conducted. Results of tensile tests at various strain rates at 650°C indicated two promising treatments in the 16-9DL alloy. Both treatments produce fine grain (~5 to 10 micron) material; one consists of 1 hour anneals at 1040 and 870°C following 60 to 80 percent cold reduction, and the other involved 930°C annealing following 50 percent reduction at −196°C. The double anneal treatment was used in conjunction with planetary swaging to produce tubes which meet [illegible] cal specifications (except length) set forth by the [illegible]. Initial creep tests of this fine grain tube material [illegible] indicated slightly higher strength and about one-third greater [illegible] ductility than normal Type 316 stainless steel tested under [illegible] conditions. Removal of carbon impurities from molybdenum [illegible] studied in moist hydrogen and argon atmospheres using an improved carbon analysis procedure. In both atmospheres, [illegible] removal appeared to follow different reactions above and below 1400°C. In hydrogen containing 20 ppM H_2O, carbon removal [illegible] rapid at the higher temperatures, in sheet specimens 0.[illegible] cm thickness, carbon concentrations of 265 ppM were re[illegible] to 2 ppM in less than 1 hour at 1600°C or higher while 1[illegible] appeared necessary for the same reduction at 12[illegible]°C. Carbon reduction was slower in moist argon than in hydrogen and at temperatures above 1400°C the residual carbon concentration after 2-hour exposures appeared related to carbon solubility. A [illegible] of variables influencing the weldability of molybdenum and W-Re-Mo alloys established that powder-metallurgy molybdenum compares favorably with a vacuum arc-cast molybdenum in terms of weld porosity and bend ductility. Optimum electron-beam welding

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Item #2

MATERIALS PROBLEMS IN STEAM GENERATORS—II: OPERATING EXPERIENCE

1. Operating Experience in Fossil-Fueled Stations, *S. Featherby (Ont-Hydro), Invited*

No summary available.

2. Operating Experiences of Steam Generators in Pressurized-Water Nuclear Plants, *W. M. Byerley (W-Phila), Invited*

The earliest operating experiences with steam generators were with such early pressurized-water plants as the BR-3 at Mol, Belgium, and Rowe Yankee, where reactor plant criticality was achieved in 1960. Since then operating experience has been collected from successively larger sized units to the present ones capable of generating steam for electrical capability in excess of 250 MW(e) each.

In the early plants, the material for the pressure boundary parts was of a carbon steel composition. As technology advanced and as the steam generator size increased, the material for pressure-containing parts was changed to the higher-strength low-alloy manganese-molybdenum steel. The tubes have likewise experienced a change. The initial units utilized type-304 stainless steel, whereas for the [illegible] steam generators Inconel [illegible] as a standard because of its resistance to chloride stress corrosion. The thermal performance has been monitored and evaluated on these plants as their size has increased to the present units now operating at San Onofre and Connecticut Yankee.

Continued surveillance has been maintained on the Rowe-Yankee and Selni steam generators to determine the effect of deterioration in heat-transfer performance through the operating life of the plant. The results of this surveillance have shown that the heat-transfer performance has not degraded as the operation of the plant has progressed.

The reliability of the steam generators for pressurized-water plants has proven to be very high. In particular, the integrity of the pressure boundary through the tubes between the primary and secondary sides of the steam generator has been a problem in our units only on the Rowe-Yankee steam generators where leakage was detected late in 1967 after seven years operation. This leakage was pinpointed in the stainless-steel tubing at approximately the secondary face of the tube plate. Twelve tubes were plugged by direct access into the primary chamber without decontamination.

As a field followup of the steam generators, samples from the CVTR steam generator were removed after five years operation, to see if stress corrosion of the secondary system surface of the Inconel tubing is occurring. Metallurgical investigations of the tubes in the tube plate as well as the tube supports are now under way.

3. Peach Bottom Stress Corrosion Cracking, *L. J. Hull, J. F. Watson (GGA), Invited*

Prior to the installation of the nuclear core in the Peach Bottom Power Plant and after a series of hot runs with this gas-cooled system, stress corrosion cracking was discovered in the type-304 stainless-steel superheater components of the steam generators. The hot runs were part of the plant checkout program and consisted of operating the system isothermally with the heat of compression from the helium circulators. After completion of the hot runs it was discovered that a type-304 stainless-steel bellows in the superheater outlet section had failed by nonductile cracking. Examination of the bellows in the other steam generator superheater section showed it had failed in the same manner. Metallographic examination of cross sections of this bellows showed extensive cracking characteristic of stress corrosion failure.

As a result of the failure in the bellows, the type-304 stainless-steel superheater tubing also became suspect. Borescopic examination of these tubes revealed no cracks, but further tests with a liquid penetrant inspection showed indications in an area below the tube-to-tube-sheet seal weld. A pressure test showed small leaks in several of the tubes below the seal weld. Sections of several tubes were cut very carefully from the superheater inlet tube sheet, including samples from the area of the tube that had been rolled into the tube sheet hole during fabrication. Metallographic examination of these lengths of tubing showed stress corrosion cracking. The cracks were associated with transition of the rolled to the nonrolled area of the tube and occurred both above and below the rolled area but not in the rolled portion.

In addition to the bellows and tubes, part of the superheater outlet was also constructed of type-304 stainless steel. Examination of this pipe showed indications of stress corrosion cracking in areas near welds and in a bend.

It is known that stress corrosion cracking can be produced in austenitic grades of stainless steel. The requisite conditions include tensile stresses and a corrosive environment (chlorides and oxygen). Elevated temperatures accelerate such cracking. The Peach Bottom steam generator cracking was attributed to the fact that stressed type-304 stainless-steel components were exposed to this environment before and during the period of the hot runs.

Tests were conducted to learn more about the nature of stress corrosion. Cracking similar to that which occurred in the steam generator was produced when simulated tube-to-tube-sheet joints fabricated like the steam generator joints were exposed to boiling magnesium chloride. Further tests verified that a tensile stress in the material was necessary to produce cracking. This suggested that exposure of type-304 stainless steel in boiling magnesium chloride could be used to indicate the presence of residual tensile stresses after fabrication. By the use of this technique a method of taper rolling was developed that showed little or no residual tensile stress on the inside surface of a tube that had been rolled in a tube-to-tube-sheet joint. The taper rolling method consisted of rolling with a tool that had long rollers set at a low angle to the axis of the tube. With this tool the tube could be rolled into firm contact with the tube sheet hole in the seal weld area and still not exhibit a step transition from a rolled to unrolled section. The diameter of the taper rolled area gradually decreased and blended into the inside diameter of the unrolled tube.

The steam generator superheater sections were retubed with Incoloy-800 (ASME SB-163, Grade 2). The higher nickel content of this material (30 to 34%) gives

Item #3

Excerpts from
WAPD-MRP-118-135
June 1966 – Jan. 1971

A. [illegible] W. J. Purcell, Manager

1. Fluid Systems and Plant Engineering (1000XP2) R. J. Devoy

This project covers the engineering effort related to PWR power plant systems and plant engineering required to support Core 2 operations at Shippingport and to accomplish any necessary modifications.

a. Fluid Systems

Valve Operating System C. H. Hays

A detailed review of the Valve Operating System was completed during this report period. The review was undertaken because of demonstrated unreliable operating performance of the PURAL System By-Pass Valve from August 6 thru 10. Operation of the PURAL By-Pass Valve caused a momentary bounce of one or two main coolant hydraulic operated stop valves and it was not certain whether the By-Pass Valve would remain in the desired position following operation.

From the review it was apparent that leakage past certain 3-way selector control valves then back to the normally depressurized water bottle was depressurizing the VOS vent header line. The vent header should be at approximately reactor coolant system pressure for reliable VOS operation. [illegible] therefore recommended, that a two-inch lift check valve be installed in the upstream VOS water supply header to block back flow out of the VOS and prevent depressurization of the vent header. This modification was approved and the check valve installed during the October Maintenance Shutdown. Initial observations have indicated that the check valve successfully prevents depressurization of the vent header.

b. Plant Engineering

1B Boiler C. H. Hays

On August 26, Fluorine-18 activity was detected in the 1B boiler (U-tube heat exchanger) steam side during a routine sample and analysis check of secondary boiler water. The activity correlated to a primary to secondary leak rate of approximately three gallons/hour. This activity in the 1B boiler was apparently due to a leak of primary water to the secondary side as no activity had been detected during a similar check on August 23.

The activity of the 1B boiler secondary held fairly constant to the levels of initial detection (80 - 100 dpm/ml) until late on September 6 or early on the 7th when the Fluorine-18 activity increased to 850 dpm/ml. The evening of the 7th the 1B loop and boiler were removed from service because of the continued high activity indicating a higher leak rate of 30 - 50 gph. Shortly after the partial isolation of the 1B loop and start of a cooldown, the 1B boiler apparently developed a significantly greater leak rate, of the order of 100 gph as determined from operating plant parameters.

Subsequent leak testing of the isolated and drained boiler revealed one tube with a leak at the mid-point of the U-bend. The tube is in the approximate center of the tube nest.

Effort is continuing to obtain a suitable inspection technique for examining the U-bend area. Eddy current tests of the straight lengths of the leaking tube and 100 other tubes showed no tubing flaws.

<u>Flushing of the Condensate Header</u> W. A. Dadd

The condensate header and the piping associated with the Safety Injection System booster pump have been flushed, on a monthly basis from each of the three flush connections, to minimize the possibility of carrying foreign material into the Safety Injection lines. Since the amount of material collected on the flush cloth during each flush as presently conducted has been sufficiently small, a recommendation was approved to reduce the frequency of flushing to once per quarter from each of the three connections. Subsequent evaluations will be made to further modify the frequency of flushing if necessary.

2. <u>Control Engineering</u> (1000XT2) (1030XM3) R. J. Devoy

This project covers the engineering design and support of the PWR Control System.

a. <u>Nuclear Protection System</u>

<u>Reactor Protection Analysis from 5000 EFPH through 10,000 EFPH</u> J. J. Cox

A complete review of the reactor protection analyses covering the period from 5000 EFPH through 10,000 EFPH was performed.

The thermal design criteria used in determining limiting core conditions are as follows:

1) Departure from Nucleate Boiling (DNB) shall not occur. The PWR DNB design equations have been used in determining allowable core capabilities.

2) No steady-state flow oscillation or transient flow reversal is permitted within the coolant channels.

3) Fuel and clad temperature limits shall not be exceeded at anytime.

4. Chemistry Applications (301B180) P. D. Brown

This project includes the evaluation of chemistry tests and operating data, consultation on chemistry problems and fission product release, and consultation and development effort pertaining to the detection and location of failed fuel elements.

a. Applications

PWR Test DICS 59405 - High pH Ammonia Test W. Lechnick

Testing is being performed to determine if core pressure drop buildup varies inversely as a function of pH below some level, or if core pressure drop buildup is a step function of pH. Starting on August 6, 1966, the pH was allowed to drift downward from a pH range of 10.1-10.3 (nominal 10.2) to a range of 9.8-10.0. Flow decrease in the diagonal clusters was about 0.125 percent per day at pH 10.1. Flow decrease in the diagonal clusters for the pH range 9.8-10.0 was on the order of 0.25 percent per day, which is the same rate noted in the summer of 1965 prior to the initiation of high pH (10.2) operation. Ten days of operation were required to establish the new buildup rate.

On August 18 the pH was raised to 10.3 (±0.1). Consistent with past experience, an immediate leveling-off in core pressure drop was observed. With continued operation at pH 10.3, there was some indication of flow recovery in the diagonal clusters. Unfortunately, the problem with the 1B boiler leak and subsequent plant shutdown interrupted the test before a definite trend had been established. After the plant startup on three loops, operation continued at pH 10.3 for about two weeks, until shutdown for test purposes, with no indication of any trend up or down in core pressure drop. However, the probability of reducing core pressure drop by pH 10.3 operation subsequent to the plant shutdown was small because of the crud knocked off the core by the reactor shutdown and check valve slams due to pump switching.

In conjunction with the core pressure drop measurements, data were obtained on flow in the FEDAL System lines during operation at pH 9.9 and 10.3. It was determined that the FEDAL System flow is very sensitive to pH. At pH 9.9 the flow in the FEDAL sample lines decreased at a rate between 2 and 13 percent per day. The valves had to be adjusted frequently to compensate for this change. At pH 10.3 there was some recovery of flow in these lines, following which flowrate remained constant.

Tritium Activity W. Lechnick

Analytical data from Shippingport indicate that primary coolant tritium activity has decreased from the peak values noted in June 1966.

TABLE A-XVI

PWR REACTOR COOLANT TRITIUM ACTIVITY

Sample Source	Seed 1 - Core 2* Sample Date	Tritium Activity (uc/1)
AC-A1X	7-21-65	6.7
BD-A1X	8-17-65	4.7
BD-A1X	10-13-65	8.8
BD-A1X	2- 3-66	12.9
AC-A1X	3-11-66	15.0
AC-A1X	4-12-66	29.3
AC-A1X	5-19-66	43.2
AC-A1X	6- 2-66	55.1
AC-A1X	7-13-66	48.0
AC-A1X	8- 1-66	45.0
AC-A1X	9- 7-66	53.0

*Tritium concentration during Core 1 operation ranged from 2-4 μc/1 following the changeover to LiOH pH control. The source of the tritium activity has not yet been established.

PWR 1B Boiler Leak W. Lechnick

Chemistry samples were examined in conjunction with the PWR 1B Boiler leak incident. No analytical evidence of secondary system chemicals in the primary coolant was detected. Na^{24} activity in the primary coolant was calculated to be somewhat higher than normally expected for PWR. On September 7, when the plant shutdown occurred, the Na^{24} activity in the 1B Boiler was 2.4×10^3 dpm/ml. This value was calculated to require between 1 and 1.5×10^4 dpm/ml Na^{24} in the primary system, corresponding to about 1 ppb of sodium in the primary coolant. Primary coolant analysis on September 14, following plant startup, showed a Na^{24} activity level of 5.25×10^3 dpm/ml, in good agreement with the range of sodium activity normally expected. No further investigation of Na^{24} activity was required.

Leak rates in the 1B Boiler were estimated from the F^{18} activity found in the boiler water. Table A-XVII tabulates the leak rates calculated.

TABLE A-XVII

LEAK RATE IN THE 1B BOILER

Date	F^{18} Activity in 1B Boiler (dpm/ml)	Power (%)	Power History[a]	Leak Rate (ml/min)[b]
8-2[illegible]-66	0	67	Steady State	0
8-2[illegible]-66	85	67	Steady State	155
8-26-66	118[c]	67	Steady State	[illegible]
8-27-66	93	67	Steady State	[illegible]
8-28-66	41	20	Fluctuating	---
8-29-66	81	67	Steady State	147
8-30-66	71	67	Steady State	129
8-31-66	87	67	Steady State	158
9- 1-66	78	67	Steady State	142
9- 2-66	118	67	Steady State	215
9- 3-66	123	67	Steady State	225
9- 4-66	46	16	Fluctuating	---
9- 5-66	23	18	Fluctuating	---
9- 6-66	46	28	Fluctuating	---
9- 7-66	858	67	Steady State	1370 = 21.8 gal/hr
9- 7-66	Leak rate suddenly increased late that evening to about 100 [illegible].			

(a) For a period of 4-6 hours prior to sampling.
(b) Estimated range of error is on the order of ±40%.
(c) Twelve hour steady-state operation prior to sampling.

PWR Test 53401 - FEDAL System Operational Test P. W. Frank

A new computer program KOSMY was completed to process data taken in DLCS-53401, "FEDAL System Operational Test". This test determines the delayed neutrons in the effluent from the 97 fuel assemblies in PWR Core 2 in order to detect any failed fuel elements. The code computes the relative activity (ratio of delayed neutron activity in the effluent of a given assembly to the average activity) for each assembly. The data are taken randomly about the core. The code arranges the relative activities into equivalent power regions of the core, gives the region, location of the assembly, monitor number, and port number. It computes the average relative activity for each region, the standard deviation, and the relative deviation for each assembly (average activity minus the individual activity divided by the region standard deviation). A report will be issued for the performance of the subject test completed during the first half of PWR-2 Seed 1 life.

A. PWR PLANT SUPPORT R. J. Devoy, Manager

1. Fluid Systems and Plant Engineering (100GXP2) R. J. Devoy

This project covers the engineering effort related to PWR power plant systems and plant engineering required to support Core 2 operations at Shippingport and to accomplish any necessary modifications.

a. Fluid Systems

Failed Element Detection and Location System (FEDAL) W. A. Budd

During a scheduled FEDAL operational checkout test (DLCS 58201), a low flow condition was noted on sample line No. 2 following the return of the two sample lines to normal service. The condition indicated a stuck-closed excess flow check valve in line No. 2. An approved backflow procedure, utilizing the low flow hydrostatic test pump was initiated to re-open the check valve. The attempt was unsuccessful and a new procedure was provided whereby a charging pump output (0 - 25 gpm controled by throttling a recirculation line around the pump) was utilized to provide higher flows than the 2 gpm of the hydrostatic test pump. The charging pump method, with only partial throttling of the recirculation line flow, successfully opened the check valve and normal one (1) gpm flow was regained in the No. 2 sample line.

b. Plant Engineering

1B Boiler D. T. Klinksick

Work continued on the inspection of the 1B boiler tubes one of which developed a leak as reported in Part I, Section A-1-b, 1B boiler of WAPD-MRP-118 covering the report period July 23, 1966 to October 21, 1966. It was determined that the failure was an opening 40 - 60 mils in width and about 1-1/4 inches long. A device was developed to detect wall thinning. Wall thinning tests were conducted on the leaker and the six tubes surrounding the leaker. The tests revealed some thinning at the failure in the leaker and no evidence of thinning was detected in the other six tubes.

The leaking tube was plugged when testing was completed and successfully passed the leak test. Each tube was then individually flushed with demineralized water and the head and primary coolant nozzles cleaned with demineralized water. Inspection of the tubes, head and primary coolant nozzles indicated that the flushing and cleaning operations were successful. The 1B boiler was returned to service on November 28, 1966 and the steam side (secondary water) was monitored for Fluorine-18 activity. No activity was detected.

On January 15, 1967, seven (7) weeks after being returned to operation, the 1B Steam Generator developed a rapid leak (estimated at 150 gpm) from primary to secondary during plant heatup. The 1B loop is now isolated for further investigation.

4. Chemistry Applications (SOFTICO) P. E. Brown

This project includes the evaluation of chemistry tests and operating data, consultation on chemistry problems and fission product release, and consultation and development effort pertaining to the detection and location of failed fuel elements.

a. Applications

Tritium Activity at PWR P. W. Frank, W. Lochnick

Analytical data from Shippingport, tabulated in Table A-III, show decreasing values from the high point noted in June of 1966. Primary coolant losses have increased significantly from values noted in June, and are partially responsible for the decrease in tritium activity. The decrease between November and December is probably not a true decrease since leak rates during this period were substantially greater than during the preceding period.

TABLE A-III

PWR REACTOR COOLANT TRITIUM ACTIVITY

Sample Source	Seed 1 Core 2* Sample Date	Tritium Activity (μc/l)
AC-A1X	6-21-66	55.1
AC-A1X	7-13-66	48
AC-A1X	8- 1-66	45
AC-A1X	9- 7-66	53
AC-A1X	11- 3-66	31
AC-A1X	12-14-66	19

*Tritium concentration during Core 1 operation ranged from 2-4 μc/l following the changeover to LiOH pH control.

There are so many nuclear reactions that produce tritium that it is not possible to establish at this time the precise source of the tritium in the coolant of PWR Core 2. Investigations conducted on the possible origin of this variation in tritium activity are discussed in Sections I.C.2.a and I.D.1.b.

DIGA-59505 - High pH Ammonia Hydroxide Test in PWR W. Lochnick

Testing is in progress to determine core pressure drop buildup at pH 10.0 ±0.1. The specified pH range was achieved on December 25, 1966, and

I: PWR ENGINEERING

A. PWR PLANT SUPPORT R. J. Devoy, Manager

1. Fluid Systems and Plant Engineering (1000XP2) R. J. Devoy

This project covers the engineering effort related to PWR power plant systems and plant engineering required to support Core 2 operations at Shippingport and to accomplish any necessary modifications.

a. 1B Steam Generator C. E. Mays

The 1B Steam Generator was shutdown, isolated and drained during this report period because of a second primary to secondary leak as reported in MRP-119.

The leak location was established initially by a secondary side water test with a moveable pneumatic stopper inserted from the primary side of the tube; the water test was then later confirmed by an ultrasonic probing of the primary side of the tube. As with the first leak reported in MRP-118 one tube was found to be leaking because of a tube failure in the same U-bend area as the first leak. The size of this second leak appears to be approximately the same as the first and is in a tube in the same vertical column as the first leaker but eight rows higher. (PWR steam generators are installed in a horizontal position.)

Efforts to date have been directed to developing: 1) a machine driven ultrasonic shear wave test for inspecting tube U-bends for partial or complete crack defects easily missed by the hand operated normal ultrasonic beam, 2) a tube cutting device capable of making through cuts on the order of 23 feet from the tube sheet for removing U-bend samples from peripheral tubes accessible from secondary side, and 3) a visual means either by camera or flexible boroscope of examining the failed area. Efforts are promising particularly in the case of UT and Tube Cutting, however, to date, a satisfactory and complete field operation has not been demonstrated and it appears another three to six weeks will be required before such a demonstration can be made.

b. Cooldown of the 1B Steam Generator L. E. Kennet

Procedural guidelines and limitations were prepared to expedite cooldown of the isolated 1B steam generator. This action was considered necessary as the natural temperature decay rate of the isolated unit was not sufficient to assure that the unit could be drained and laid up for maintenance by the desired date. The recommended steam generator cooldown mode consisted essentially of a series of partial draining and filling operations that would add cooler water periodically to the secondary side of the steam generator. Installed temperature instrumentation will be monitored to assure compliance with existing plant thermal limitations.

c. Heatup and Cooldown Operations L. H. Kemmet

A recommendation was approved to reduce plant pressures prior to initiating cooldown operations and maintain reduced pressures until heatup operations are completed. This mode of operation would decrease the primary-to-secondary pressure differential across the steam generator tube walls. Essentially the plant pressure would follow the minimum pressure set by the reactor coolant pump operation curve during the heatup and cooldown operations. It is noted that both of the 1B steam generator tube ruptures occurred during a cooldown and heatup operation, respectively, when primary-to-secondary pressure differentials were at their highest values.

d. 1B Steam Generator Decontamination W. A. Budd

An electrolytic decontamination of the primary side inlet and outlet plenum of the 1B steam generator was accomplished to minimize radiation exposure during planned inspection of the steam generator tubes. Only the plenum walls and divider plate area were decontaminated; the steam generator tubesheet was protected from the decontamination solution by a polyethylene sheet.

The electrolytic decontamination is a deplating operation using dilute sulfuric acid as the electrolyte at the point of contact. A special brush was developed to apply the electrolyte at a controlled rate to the plenum walls and to minimize runoff All of the decontaminating on the plenum inlet side was performed from the outside using an extension handle on the brush. On the outlet side plenum, the operator had to enter the plenum in order to effect satisfactory results. A maximum of 20 volts DC, with current varying between 10-30 amp depending on rate of deplating, was utilized. A total of about eight gallons of electrolyte was needed. The entire operation went without incident.

The decontaminated sections had the appearance of bright shiny metal compared to a dull black before the operation. Autoradiographs will be taken of the decontaminated areas for comparison with those taken prior to the descaling operation. The results will be reported in the next issue of the MRP.

e. Two Loop Power Operation L. H. Kemmet

The 1B steam generator is presently out of service to repair a primary-to-secondary leak and, in the event another loop must be removed from service, continued power operation could only proceed with two loops in service. A review showed that satisfactory power operation can be achieved at Shippingport with only two operating reactor coolant loops provided a coolant purification loop is maintained in service. In the unlikely event that neither purification loop would be available, piping modifications would be required to fulfill existing coolant chemistry sampling requirements and provide for hydrogen addition. The maximum power level with two loop operation is 45%.

4. Chemistry Applications (301080) P. E. Brown

This project includes the evaluation of chemistry tests and operating data. consultation on chemistry problems and fission product release, and consultation and development effort pertaining to the detection and location of failed fuel elements.

a. Applications

Tritium Activity at PWR W. Lechnick, P. W. Frank

Analytical data from Shippingport, tabulated in Table A-I, indicate tritium activity has leveled-off after decreasing from the high point noted in June 1966. The low value noted in December 1966 can be attributed to above-average leak rates preceding this sampling date. The level of tritium activity found has presented no operational or disposal problems.

TABLE A-I

PWR REACTOR COOLANT TRITIUM ACTIVITY

Sample Source	Seed 1 Core 2 * Sample Date	Tritium Activity μc/1
AC-AIX	6-21-66	55.1
AC-AIX	7-13-66	48
AC-AIX	8-1-66	45
AC-AIX	9-7-66	33
AC-AIX	11-3-66	31
AC-AIX	12-14-66	19
AC-BIX	1-24-67	26.8
AC-BIX	2-6-67	22.9
AC-AIX	3-6-67	26

* Tritium concentration during Core 1 operations ranged from 2-4 μc/1 following the changeover to Li^7OH pH control.

There is little or no holdup of tritium on the ion exchange resin. The effect of decay (12.26 year half life) is negligible in establishing the steady state tritium concentration which exists. Plant coolant "leakage" is responsible for the major removal of tritium activity from the system. Since tritium concentration is relatively constant in the range of 25 uc/l, and the "leak" rate is approximately 25 gph, the production rate of tritium can be estimated as follows:

$$25 \text{ uc/l} \times 3.78 \text{ l/gal} \times 25 \text{ gph} = 2.4 \times 10^3 \text{ uc/hr.}$$

Possible tritium production mechanisms were discussed in WAPD-MRP-119, D.1.b. A production rate of 2.4×10^3 μc/hr. does not enable elimination of any of those production mechanisms from consideration.

BLCS-59505 - High pH Ammonium Hydroxide Test in PWR W. Lechnick

A rate determination was achieved for core pressure drop buildup at pH 10.0 ± 0.1. Core pressure drop measurements have now been made for pH 10.2, 10.1 and 9.9 operation. The data are summarized in Table A-II.

TABLE A-II

PWR CORE PRESSURE DROP AND FLOW AS A FUNCTION OF pH

	pH *			
	9.9	10.0	10.1	10.2
Core Region	All values in psid per 24 hour day			
Seed ΔP	+0.23	+0.1	+0.08	No change
Blanket ΔP	+0.11	+0.06	+0.04	No change
Seed Fuel Assemblies	All values are in % flow reduction per 24 hour day with 100% being the hot, 4 loop, pumps on fast, design flow.			
Center		-0.01		No change
Diagonal	-0.25	-0.03	-0.04	No change
Inset		-0.05		No change

* All pH values are ± 0.1

I: PWR ENGINEERING

WAP-121

A. PWR PLANT SUPPORT — R. J. Devoy, Manager

1. Fluid Systems and Plant Engineering (1000MP2) — R. J. Devoy

This project covers the engineering effort related to PWR power plant systems and plant engineering required to support Core 2 operations at the Shippingport Atomic Power Station and to accomplish any necessary modifications.

a. B Steam Generator — C. H. Mays

The B Steam Generator remained shutdown during this report period because of a primary to secondary tube leak as reported in WAPD-MRP-120.

Ultrasonic Testing of approximately 400 tubes using a normal incidence wave technique was conducted and completed during this period. Wall thinning in the U-bend area was found in 32 tubes which have their U-bend terminus in contact with the last tube support assembly. Thirty of these tubes with a wall thinning in excess of 40% have been plugged while the 2 tubes with less than 40% wall thinning have been left open for reinspection of tube walls following several months of B loop operation. The leaking tube and the one tube that would not accept the UT probe in the U-bend area were also plugged. Return of the B unit to service was in progress at the end of this report period.

The UT inspection was conducted using a machine drive device for moving the UT probe and wand in the tube. Signal record traces were obtained on all tubes inspected to permit comparison with future inspection data.

b. C Steam Generator — C. H. Mays

A small primary-to-secondary leak was detected in the C Steam Generator on May 18, in the course of routine daily activity analysis of secondary boiler water. The leak rate, initially about 3 DPM/ml Fluorine-18 equivalent to 0.1-0.2 gph, has steadily increased the past two months and by the end of this report period was in the range of 200 DPM/ml Fluorine-18 equivalent to 8-9 gph.

The Shippingport Atomic Power Station has been operated mainly at a base load of 100 MW(e) since the observance of the C leak, rather than in the normal swing load mode, to minimize plant temperature and pressure variations that could adversely effect the C loop leak rate.

When the B loop is returned to service, the C loop will be removed from service for inspection and repair in accordance with the program to be established by the Westinghouse Plant Apparatus Division.

MRP-121

4) Resin has a saturation level for retention of crud. Having reached saturation capacity, the resin will retain essentially no additional crud. The saturation level for PWR operating conditions is not known.

5) Mechanical or hydraulic shock, such as that resulting from pump switching operations, can contribute to loss of crud from the resin. Further testing is required to determine the desirability of momentarily interrupting purification flow for the loop involved in pump switching operations.

6) Survey data indicate that complete removal of activity is not effected with discharge of resin from the demineralizer.

7) Further testing is required to determine the effect of hydraulic shock on radiation levels in the demineralizers and to establish resin discharge criteria.

8) To minimize activity holdup within demineralizer internals, the demineralizer should be purged with forward flush after resin discharge.

Tritium Activity at PWR - W. Lechnick

Tritium activity levels for April, May, and June are 20, 29, and 37 uc/l, respectively. Although these levels appear to show an increasing trend, this is not considered significant at the present time. The average level for April, May, and June is ~29 uc/l as compared with an average of ~25 uc/l for January, February, and March, 1967.

Evaluation of Tests 58101, 58001, 58401, and 58301 - P. W. Frank

Test evaluations were submitted for the following:

DLCS-58101, "Determination of Radionuclide Base Levels,"
DLCS-58001, "Reactor Coolant Fission Product Activity,"
DLCS-58401, "FEDAL System Operational Test," and
DLCS-58301, "FEDAL System Operation During Station Startup".

The evaluations concluded that none of the tests gave an indication of the existence of fuel element defects.

5. Plant Materials (300F3B1, 307D3B0) A. W. Klein

This project covers the inspection and evaluation of parts and components removed during plant modifications, and consultation on component fabrication or installation problems, effect of unusual conditions such as decontamination solutions on plant materials, and selection of materials and fabrication processes.

I: PWR ENGINEERING

A. PWR PLANT SUPPORT R. J. Devoy, Manager

1. Fluid Systems and Plant Engineering (1000XP2) R. J. Devoy

This project covers the engineering effort related to PWR power plant systems and plant engineering required to support Core 2 operations at the Shippingport Atomic Power Station and to accomplish any necessary modifications.

a. B Steam Generator C. E. Mays

The B Steam Generator was returned to service on July 22. Current plans are to reinspect the B heat exchanger at a later date to investigate the rate of tube wall thinning. Two tubes that had shown a small amount of wall thinning were left unplugged to aid the investigation of wall thinning rate.

b. C Steam Generator C. E. Mays

The C Steam Generator was removed from service on July 22 and prepared for tube leak testing and ultrasonic inspection. One tube was found with a leak in the apex area and 14 tubes were found to have wall thinning in the apex area. As in the case of the B Steam Generator the leaking tube and those with wall thinning all had their "U" bend apex touching the loop end tube support.

The 14 tubes with wall thinning and the one leaking tube were plugged and the unit returned to service on October 20.

c. Air Cooling System W. A. Budd

A periodic inspection program on high efficiency particulate filters in ventilation exhaust systems will be instituted at Shippingport. Each filter will be tested following its installation and filters not replaced at least once between refueling periods will be tested prior to the refueling period. The inspection program will utilize the dioctyl phthalate (DOP) test method.

d. Radioactive Waste Disposal System W. A. Budd

Pursuant to the approval of modifications for improving waste disposal operations (outlined in WAPD-MTP-121, Section I.A.1.d.) job orders were prepared and submitted to the AEC for installation of a filter around chemical waste tanks, installation of an alternate distillate line from evaporator to test tanks, and installation of submicron size ion exchanger pre-filter units.

c. Radioactive Waste Disposal System (RWDS) W. A. Budd

Flooding of Surge Tank Enclosures

Each of the four RWDS underground surge tanks is situated underground in a circular pit which is maintained dry by periodic eduction of associated sumps. Due to a leaking steam valve in the B surge tank pit and a failure of the normal pump-out system, the water level between the pit concrete wall and the surge tank rose to such a level that the tank floated. The annular space water was pumped out to allow the surge tank to resettle. A visual examination revealed that the tank and its connecting lines were undamaged except for two thermocouple lines above the top of the tank. Since these lines terminate in wells inside the tank, the tank integrity was not impaired. The conclusion from the examination was that the tank was not damaged and could be returned to service.

RWDS Cooling Water pH

The pH control range for the waste disposal cooling water system was lowered from a range of 8.3-10.5 to a range of 8.3-8.8. The lower range should offer better corrosion protection to the newly installed aluminum housing around the cooling tower, and should provide complete corrosion protection for the carbon steel components of the cooling water system.

d. Heat Dissipation System Main Steam Motor Operated Stop Valve D. T. Klinksiek

A new gate valve was installed during a plant shutdown in February to replace the leaking heat dissipation main steam stop valve. This new valve is a parallel disc gate valve developed under WPAD cognizance for use in secondary side steam systems. The installation was made to replace the leaking valve, and provides an opportunity to obtain operational data on the new valve for design evaluation purposes.

e. 1B Coolant Loop Steam Generator D. T. Klinksiek

Ultrasonic testing of the steam generator tubes was started in March to study wall thinning rate at the U-bend apex. All previously ultrasonicly tested unplugged tubes were re-examined in the area where the U-bend apex is in contact with the loop end tube support. Two of these tubes had been previously reported in WAPD-MRP-122 to have a small amount of wall thinning at the apex area. Re-examination revealed increased wall thinning at the apex area. Two additional tubes, which previously had no wall thinning indications, indicated a small amount of wall thinning in the apex area.

Wall thinning inspection was also conducted on these previously tested tubes at the 4th and 5th tube support plate areas as referenced from the tube sheet. No wall thinning was indicated.

I: PWR ENGINEERING

A. PWR PLANT SUPPORT R. J. Devey, Manager

1. Fluid Systems and Plant Engineering (1000XP2) R. J. Devey

This project covers the engineering effort related to PWR power plant systems and plant engineering required to support Core 2 operations at the Shippingport Atomic Power Station and to accomplish any necessary modifications.

a. 1B Coolant Loop Steam Generator D. T. Klinksiek

A program for ultrasonic reinspection of the 1B loop Steam Generator was undertaken during the previous report period, as described in WAPD-MRP-124, Section I.A.1.c. The program revealed two cases of additional thinning over and above that identified in mid-1967 (cf WAPD-MRP-121, Section I.A.1.a.). It has been concluded that the tube thinning mechanism is a slowly progressing function which is not seen outside a relatively limited area wherein U-tube ends contact the last tube support plate. No indications of wall thinning were obtained from inspection of selected tubes at other tube support locations.

Plugging of five tubes was completed during this period. The plugged tubes included the two tubes which had been found to incur thinning during the June 1967 testing but were left unplugged (cf WAPD-MRP-121), the two additional tubes in which wall thinning was observed during the present testing, and one tube in the suspect region which could not accept the ultrasonic test equipment.

Four steam generator tubes which were plugged initially in June 1967 were unplugged, ultrasonically reinspected, and replugged. This inspection revealed no detectable additional thinning had occurred in these tubes during the period they were plugged. Based on this finding, a course of action is being pursued which will stop flow in all tubes located in the suspect area, and thus avoid further deterioration of tube integrity. The flow blocking device will be installed during the next report period.

b. Heat Dissipation System Main Steam Motor Operated Stop Valve
D. T. Klinksiek

The new gate valve, installed during the plant shutdown in February 1968 (WAPD-MRP-124, I.A.1.d.), continued to leak in excess of the 500 cc/hr maximum specified design leak rate.

The valve is installed with the stem inclined at 10 degrees below the horizontal. This stem position promotes the collection of condensate in the bonnet which can lead to thermal distortion of the discs.

WPAD recommended the installation of a bonnet drain which was approved by the AEC. A Job Order was written and the bonnet drain installed during a plant shutdown in June.

Leakage rates are now within specification requirements.

I: PWR ENGINEERING

A. PWR PLANT SUPPORT   R. J. Devey, Manager

1. Fluid Systems and Plant Engineering (10GXP2)   R. J. Devey

This project covers the engineering effort related to PWR power plant systems and plant engineering required to support Core 2 operations at the Shippingport Atomic Power Station and to accomplish any necessary modifications.

a. 1B Coolant Loop Steam Generator   D. T. Klinksiek

The effort associated with installing the flow blocking device (see WAPD-MRP-124) in the 1B loop steam generator and returning the unit to service was completed during this report period. Operation of this unit with the flow blocking device installed has not indicated any unexpected or worsening trends in plant conditions.

b. 1B Steam Generator Reduced Flow Studies   D. T. Klinksiek

Studies to determine the effects on plant operations resulting from the installation of a flow blockage device in the B loop steam generator were completed. These studies were conducted using a digital computer simulation of the reactor plant to determine steady state conditions for given plant arrangements.

The removal from service of 11.2% of the total tubes in the B unit heat exchanger by a flow blockage device was predicted to cause a total core primary flow reduction of approximately .20% from the normal plant flow with four loop operation. This primary flow reduction is for operations with four Core 2 Main Coolant Pumps installed or with three Core 2 Main Coolant Pumps installed and one Core 1 Main Coolant Pump installed in the B loop. For three loop or two loop plant operation, the total core primary flow reduction predicted is .49% and .89% respectively with similar arrangements of Main Coolant Pumps as described above. This is the only plant parameter that the studies predicted would have an influence on core power capability. The predicted deviation of other plant parameters does not indicate trends that would restrict or penalize core power operation.

These studies also included the effects on plant operations resulting from installation of a flow blockage device in the B and C units. These studies indicated trends similar to those previously discussed, only with an additional reduction of primary flow through the core. A tabulation of predicted core primary flow reductions for various plant conditions is shown below.

I: PWR ENGINEERING

A. PWR PLANT SUPPORT   R. J. Devoy, Manager

1. Fluid Systems and Plant Engineering (10OCXP2) R. J. Devoy

This project covers the engineering effort related to PWR power plant systems and plant engineering required to support Core 2 operations at the Shippingport Atomic Power Station and to accomplish any necessary modifications.

a. Radioactive Waste Disposal System   W. A. Budd

The maximum limit of vent gas activity that can be discharged from the RWDS directed through the stack is 3×10^{-7} $\mu c/cc$. Since isolated portions of the vent gas system cannot be directed to the stack, maintenance on these sections often have to be delayed to allow radioactive decay of the contained gases to the above activity level. A Bettis evaluation was approved which would allow vent gases from isolated sections to be discharged within the following considerations:

1) The volume of gas to be discharged should be as small as possible (i.e. only that section of the system necessary for maintenance should be opened to the atmosphere). The rate of discharge when opening a section should be as slow as possible.

2) In order to afford as much air dilution as can be obtained, a high capacity blower should be directed onto the vent while discharging to the atmosphere.

3) The maximum vent gas activity, prior to discharge to atmosphere, should be 1×10^{-5} $\mu c/cc$. Thus, even if only 50% of the blower capacity is realized, adequate dilution will be obtained to meet the 3×10^{-7} $\mu c/cc$ limit on discharged activity.

b. 1D Steam Generator Leak   W. A. Budd

A primary to secondary leak of 1.2 to 1.6 gallons per hour was detected in the 1D steam generator (Foster Wheeler, straight-through unit) on August 8, 1969. The unit was kept in service through September 1969 for normal plant swing load operation and also for a test period at the maximum 150 MW plant output. No significant increase in leakage occurred during any of these operations and the leak rate was about 2 gallons per hour when loop cooldown was commenced, on September 27, in preparation for the leak repair.

The D unit was opened and a secondary to primary hydrostatic test revealed a tube leak at the location of a lateral support plate. Further efforts to determine the cause of the apparent tube failure through the use of eddy current testing and, if necessary, tube removal will be made during the next report period.

I: PWR ENGINEERING

A. POWER PLANT ENGINEERING R. E. Burkhart, Manager

1. Fluid Systems and Plant Engineering (100CMP2) W. Schwartz

This project covers the engineering design and support of the PWR fluid systems and plant engineering.

a. Radioactive Waste Disposal System (RWDS) W. A. Dudd

The Shippingport Atomic Power Station has maintained a continuing record of strict compliance with all Federal and State regulations involving radioactive waste discharge. However, in compliance with the AEC policy to minimize the amount of radioactivity discharged to the environment, Bettis submitted an evaluation to upgrade the liquid process system. Approval was received to collect appropriate sludge, swipe, and liquid samples from various streams. Based on sample results and liquid waste stream flowrates, Bettis will develop a course of action to upgrade the PWR radioactive liquid waste process system operating procedures. Bettis will also develop long range measures to reduce liquid radioactivity discharged when large volumes of waste are generated.

b. Pressurizer and Pressure Relief System W. A. Dudd

Approval was received to modify the PWR primary relief valve maintenance criteria. Water relief valves that have long blowdowns and must be isolated or gagged prior to reseat will be returned to service long enough to confirm that the valve is not leaking excessively. If binding is suspected, the valve will be retested. For steam relief valves, a retest will be required if the valves do not reseat at 1700 psi, a value which is 100 psi lower than the former 1800 psi limit.

c. 1D Steam Generator Leak M. J. Hansen

Early in August of 1969, a leak was detected in a tube of the 1D coolant loop steam generator installed at the Shippingport Atomic Power Station (see WAPD-MRP-130). The steam generator has been in operation since 1965 with the exception of normal plant shutdowns, refueling and plant modifications (estimated as approximately five months of total down time). The leak was detected by the sensing of the discharge of radioactive water from the primary to the secondary side of the system. The leaking tube was confirmed, by both hydrostatic and eddy current testing (ET), to be located in one tube 1/2 up the tube bundle and in the second clockwise quadrant of the tube sheet looking from the inlet end of the generator. The tube was identified to be numbered 2647. The leaking area of the tube was located in the first of eleven tube supports from the inlet end of the generator.

The steam generator tube which leaked plus one other which ET inspection identified as being damaged were removed from the unit for detail examinations at Bettis. The inspection of the steam generator and investigation to determine the cause of the tube leak will continue during the next report period.

I: PWR ENGINEERING

A. POWER PLANT ENGINEERING R. E. Burkhart, Manager

1. Fluid Systems and Plant Engineering (100GMP2) W. Schwartz

This project covers the engineering design and support of the PWR fluid systems and plant engineering.

a. Reactor Plant Air Cooling System W. A. Dudd

Difficulties have been encountered in meeting the leak test criteria for the as-found condition for the reactor plant container air cooling system 48-inch diameter butterfly valves. Replacement of the seats and hub seals is being undertaken which will require the removal of a set of two valves from the air cooling system piping. Such removal constitutes a loss of Reactor Plant Container integrity which must be re-established upon their reinstallation.

A reinstallation procedure for the butterfly valves was approved wherein the valve piping flanges will be torqued to the same value of 515 ± 20 ft. lb as presently approved for reinstallation of removed hatch covers from the container. A leak test will be performed by pressurizing the space between the pair of valves for confirmation of leak tightness of the piping flanges between the valves. The flange nearest the Reactor Plant Container will be left torqued as nearly as possible to the condition at the flanges between the valves. The successful leak testing of the flanges between the valves will constitute acceptance of adequate leak tightness on the container side flange and eliminate the necessity of a Reactor Plant Container Leak Rate Test.

b. 1D Steam Generator Leak L. H. Kemmet

Early in August of 1969, a leak was detected in a tube of the 1D coolant loop steam generator installed at the Shippingport Atomic Power Station (see WAPD-MRP-130) and was subsequently inspected (see WAPD-MRP-131).

A program for inspection and repair for the 1D steam generator was approved and is in progress. This program includes:

1) Up to seven additional tubes are to be removed from the unit for detailed examination by the Bettis Laboratory.

2) Selected tubes with defects are to be plugged at the inlet and outlet tube sheets.

3) Special instrumentation is to be installed (see Section I.A.2.a.).

I: PWR ENGINEERING

A. POWER PLANT ENGINEERING   R. E. Burkhart, Manager

1. Fluid Systems and Plant Engineering   (1000MIP)   W. Schwartz

This project covers the engineering design and support of the PWR fluid systems and plant engineering.

a. Reactor Plant Air Cooling System   W. A. Budd

The reactor plant container air cooling system exhaust butterfly valves were removed, refurbished, and reinstalled during this period (see WAPD-MRP-132). The subsequent leak test of the cavity between the butterfly valves demonstrated that the butterfly valves cavity leakage was less than the allowable leakage.

b. 1D Steam Generator   W. A. Budd

A holding fixture for corrosion coupons was installed on the feedwater piping in the steam drum of the 1D steam generator. The fixture will hold nine corrosion coupons; three each of stainless steel, carbon steel, and NiCrFe Alloy 600. The corrosion coupons will be periodically removed and examined during future inspections of the 1D steam generator.

c. Heat Dissipation System   W. A. Budd

The two-inch line bypassing the heat dissipation system isolation valve (see WAPD-MRP-124, 127 and 129) was cut and capped during this period to determine if this line was the path for the leakage occurring downstream of the isolation valve. This modification resulted in zero leakage being measured downstream of the isolation valve. An internal bypass arrangement exists in this isolation valve which is adequate to warmup the heat dissipation system piping and components downstream of the isolation valve; the two-inch bypass line around the isolation valve can therefore remain capped.

The inspection of the steam generator and investigation to determine the cause of the tube leak will continue during the next report period. When the inspections and the above noted maintenance program are completed the 1D reactor coolant loop will be returned to service.

2. Instrumentation and Control (100GIT2) M. L. Rhodes

This project covers the engineering design and support of the PWR control systems.

a. 1D Steam Generator Special Instrumentation W. C. Thomas

Investigation into possible causes for the leak in the 1D steam generator indicates additional information is required on environmental conditions, such as thermal, hydraulic, and vibration parameters. Bettis has recommended the installation of special instrumentation on the PWR 1D steam generator to obtain data in addition to the data provided by the normal steam generator instruments. The special instrumentation installation includes thermocouples on the steam generator risers and downcomers and on the heat exchanger shell. In addition, two accelerometers will be mounted on the heat exchanger shell to provide vibration information.

The recommendation to install the test thermocouples and accelerometers was approved and a detailed description of the instrumentation installation was forwarded by Bettis.

3. Chemistry Applications (3013180) P. E. Brown

a. DLCS 58502 - Resin Discharge Criteria

This test uses average radiation survey readings near the bottom of the purification demineralizer as an indication of potential crud breakthrough and, therefore, the need for resin replacement. Latest testing, comparing the instrument presently in use with an Eberline instrument, showed that the present instrument results were erratic while consistent results were obtained with the Eberline instrument. Future surveys will utilize the Eberline instrument.

The radiation survey data also indicated the following:

1) the Eberline instrument produces higher readings than the present instrument,

2) the peak activity in the ion exchanger is normally at a point that is above the three points presently used as the criteria for resin discharge.

I. PWR ENGINEERING

A. <u>POWER PLANT ENGINEERING</u>   R. J. Devoy, Manager

1. <u>Fluid Systems and Plant Engineering</u> (10CW72)   W. Schraun

This project covers the engineering design and support of the PWR fluid systems and plant engineering.

a. <u>Primary to Secondary Steam Generator Leak Rates</u>   W. A. [illegible]

Shippingport has experienced steam generator primary to secondary leakage during PWR Core 2 operation. Subsequent to [illegible], the steam generator has been removed from service, inspected, [illegible], and returned to service. The Bettis Laboratory submitted to the AEC [illegible] during this report period an evaluation of the effect that various steam generator leak rates would have on continued plant operation. The effect of continued operation with a leaking steam generator was evaluated with respect to radioactive waste disposal system processing capability, contamination of the secondary plant, and eventual discharge of [illegible] to the environment from both normal and postulated accident conditions. This evaluation provides guidance as to when a steam generator with primary to secondary leakage should be removed from service for maintenance.

Oct. 24, 1970 -
Jan. 25, 1971