INEEL Oversight
Idaho's eyes on the INEEL

Dirk Kempthorne, Governor
Kathleen E. Trever, Coordinator

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October 26, 2001

50-397

Jack Cushing
Office of Nuclear Reactor Regulation
Nuclear Regulatory Commission
Mail Stop 07E1
1White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

Dear Mr. Cushing:

Per our telephone conversation, enclosed are materials forwarded to Governor Kempthorne by Robert Leyse, a constituent with decades of experience in the nuclear industry. Mr. Leyse is concerned that a Licensee Event Report for the River Bend reactor in Louisiana, describing conditions of crud buildup on reactor fuel, may not have received widespread attention. I could not find any enforcement action against River Bend, and Mr. Leyse wishes to ensure that the regulators of Energy Northwest Columbia Generating Station nuclear power plant in Richland are aware of this Licensee Report.

Would you please write to Mr. Leyse to let him know the extent to which the NRC has evaluated the issues of crud buildup from this Licensee report for other operating commercial boiling water reactors, especially the Richland reactor? Mr. Leyse has separately filed a petition with the NRC, PRM-50-73, to propose changes to 10 CFR § 50.46, but I would appreciate your giving him an update on any reactor-specific assessments.

In addition to responding to a constituent's concern, we also want to ensure safe operations of the Richland reactor. As such, please send me a copy of your response to Mr. Leyse to determine whether the State of Idaho needs to do any additional follow up.

Thank you for your prompt attention to this matter.

Sincerely,

Kathleen E. Trever

Kathleen E. Trever
Coordinator for INEEL Issues

KET/ds
Enclosure

cc: Dick Cowley, Washington Division of Health (w/encl.)
Doug Walker, State of Idaho NRC Liaison (w/encl.)

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UNDER REVIEW

SCALE FORMATION IN AN EARLY NUCLEAR POWERED STEAM GENERATOR

Robert H. Leyse

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BSTRACT

The Experimental Boiling Water Reactor (EBWR) was uilt and operated at the Argonne National Laboratory near Chicago during the late 1950's and early 1960's. The initial ower level was 20 megawatts. The operating pressure was 0 atmospheres and the expected surface temperature of the lat plate nuclear fuel elements was in the range of 255 egrees centigrade over a wide range of heat fluxes. However, plans to operate the EBWR at substantially higher ower levels were significantly impacted when significant cale deposits were discovered on the nuclear fuel elements. Scale deposits were most pronounced in the central regions of the reactor core where the maximum heat flux was in the ange of 500 kilowatts per square meter. These deposits were mainly aluminum oxide that was exfoliated from llegedly corrosion resistant aluminum alloy that was ncorporated in certain structures of the nuclear reactor core. The scale was extremely adherent to the zirconium heat ransfer surfaces until the thickness reached the range of 0.013 centimeters, at which point some of the scale flaked off and entered the flow of boiling water. Apparatus was assembled to measure the thermal conductivity of the scale; the value was determined to be in the range of 0.008 watts per centimeter-degree centigrade. Recently it has been reported that some fuel rods in today's commercial nuclear power plants have accumulated nearly 17% of zirconium alloy cladding oxidation during normal operation. Inz, Incorporated is now evaluating the impact of this thermal resistance on normal and also off-normal plant performance.

INTRODUCTION

The Experimental Boiling Water Reactor (EBWR) was built and operated at the Argonne National Laboratory near Chicago during the late 1950's and early 1960's. The EBWR was primarily an experimental plant and it provided a substantial base of information that has been utilized in the design and impressive operation of large central station boiling water reactors during the past several decades. Unfortunately, the experimental program was limited by thick scale deposits that formed on the flat plate fuel elements that were the primary heat transfer surfaces. This paper presents the discovery of the scale deposits, the initial denial that scale could limit plant operations, the real impact of the scale in limiting plant operations, and the results of field and laboratory work in classifying the scale. This paper also briefly speculates on the potential impact of scale on the destructive accident of a small power reactor then under development for arctic or other remote applications. Finally, the potential implications of zirconium oxide formation in today's large power reactors are discussed.

The EBWR Core

Fuel elements. The fuel elements are rectangular, boxlike structures, 197 cm long and 9.5 cm square. The components, Fig. 1, are the lower locating end fitting, the 6 fuel plates, the side plates and the top handling fitting. The

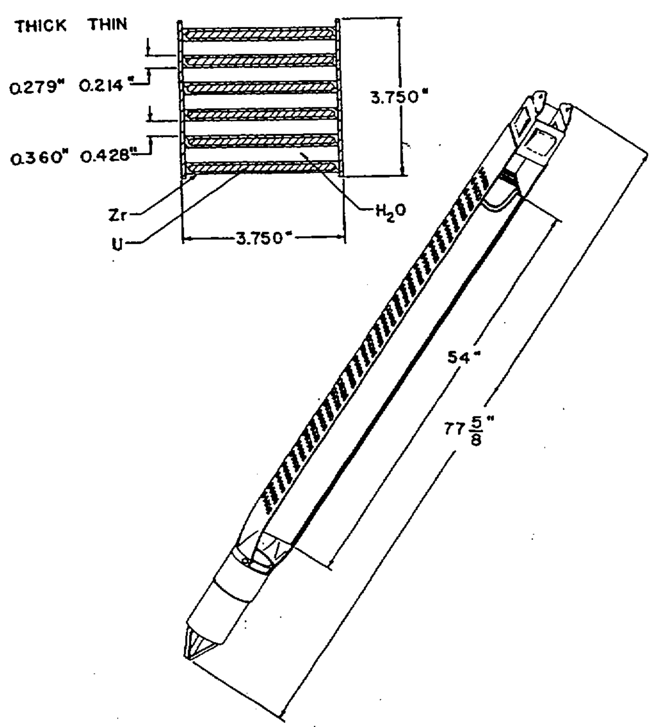

Fig. 1 EBWR fuel element. The dimensions in SI units are in the text. The units of the dimensions in this figure represent the fabrication specifications.

fuel plates are Zircaloy 2-clad uranium alloy plates, manufactured in 2 thicknesses. One plate is 137 cm long, 9.2 cm wide and 5.4 mm thick. The other plate has the same length and width dimensions, but is 7.1 mm thick. The nominal cladding thicknesses are 5 mm over the face of the uranium plates, 3 mm over the edges and 3.8 cm over the ends. This yields an effective heat transfer area of about 1100 cm^2 per face.

The side plates which measure 156 cm long and 9.5 cm wide are fabricated of Zircaloy 2 sheets. The lower end of each sheet is formed to effect a smooth transition with the lower end fitting. The purpose of the perforations, Fig. 1, is to reduce the longitudinal tensile strength of the side plate so that differential thermal expansion and/or radiation damage growth of the fuel plates can be absorbed by the assembly with minimum restraint to the fuel plate. This is a significant feature when operating with highly scaled heat transfer surfaces and this will be discussed in those sections of this paper.

The lower end fitting supports the fuel element within the core structure and it serves as the water inlet to the fuel plates.

The top fuel handling fitting facilitates remote gripping, lifting and lowering of the fuel element with tools that are manipulated from above through several meters of water that provide radiation shielding.

The reactor core assembly. The reactor core assembly includes 114 fuel elements arranged in a circular pattern. See the plan view in Fig. 2. The fuel elements are mounted on a grid plate within the pressure vessel. Nine control blade assemblies, identified as control cross in Fig. 2, are arranged in a symmetrical box pattern within the assembly of fuel elements.

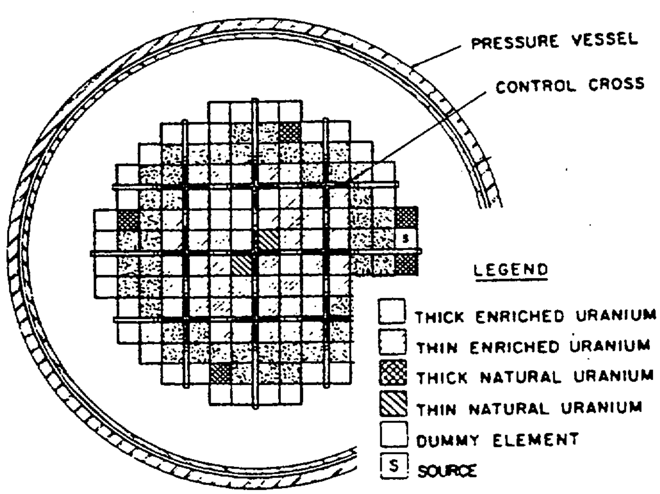

Fig. 2 Plan view of the EBWR core arrangement

Dummy elements are used in the initial core loading to fill in fuel positions in the outer ring of the core. Like the fuel elements, these dummies include a lower end fitting, a square body and a top fitting for handling. The square body of the dummy is fabricated of aluminum- 1 weight-percent nickel alloy sheet (1.6 mm thick). In the final design report for the EBWR, Soppet (1957) wrote, "Aluminum-nickel alloy is used in lieu of zirconium for reasons of economy. The alloy has exhibited satisfactory corrosion resistance in laboratory tests; however, if corrosion is excessive, the elements can be removed and replaced." This measure proved to be very costly because it limited the operational flexibility of the EBWR.

In the second phase of operation of the EBWR, the aluminum dummies were removed. The available positions were needed for additional fuel elements for higher power operation. However, only very limited high power operation was achieved. The highly scaled condition of most of the fuel elements significantly impacted the test program.

Reference heat transfer and flow characteristics. The reactor of the EBWR has the same function as the boiler of a conventional fuel fired system, i. e., the generation of steam. Cold feed water at 43 C is pumped into the unit and its temperature is increased to saturation (254 C at 41 atm). Evaporation occurs and an equal quantity of saturated steam is produced. At 20,000 kW the corresponding steam generation rate is 27,000 kg/hr.

Table 1. Heat Transfer and Flow

Parameter	Value
Reactor Power	20,000 kW
Pressure	41 atm
Steam flow	27,000 kg/hr
Average power density based on coolant within heated channels	25.3 kW/liter
Core average heat flux	143 kW/m^2
Core maximum heat flux	488 kW/m^2
Average voids at core exit	28.5 percent
Coolant flow area in fuel zone	0.78 m^2
Coolant inlet velocity in fuel cluster	0.75 m/s

FIELD EXPERIENCE WITH SCALE DEPOSITS

Summary of the Field Experience

The reactor experimental program had two phases, the first phase, December, 1956 through mid 1959, was with the reactor core assembly described above. The second phase, 1960 through 1962, was aimed at higher power operation. In the second phase, the aluminum dummy elements were removed and the space was used by additional fuel elements. The impact of the fuel element scale on operations is best summarized in the following paragraphs by Wimunc (1963).

The original EBWR core contained 114 fuel assemblies *surrounded by 32 aluminum dummy fuel assemblies to fill in the grid plate. This core remained essentially undisturbed for about 3 years in the reactor vessel. During this time, deposits varying in thickness (up to about 0.008 in.) had built up on the fuel plate surfaces. Attempts to remove the deposit proved not practical. Analysis of sample scrapings revealed a composition of 67 w/o boehmite ($Al_2O_3H_2O$), 25 w/o nickel and 8 w/o iron. Consequently, the aluminum dummy assemblies were adjudged the principal source of the scale. The exact mechanism of deposition has not been resolved, however, it is fairly well established that the mass transfer is from the dummy fuel boxes to the coolant and then to the active fuel plates, primarily on the boiling areas of the plate surfaces. All aluminum dummy fuel elements were removed from the core in order to provide space for the additional spike fuel assemblies of the high power runs.*

This scale was a matter of deep concern during operation at elevated powers. The existing scale on the fuel plates represented a barrier to effective heat transfer. The thermal conductivity of the scale has been estimated to be less than 1 Btu/(hr)(ft^2)(F)/ft). At elevated powers, the scale could promote a central fuel temperature of ~ 1125 degrees F. At this temperature , the creep strength of the uranium ally would be reduced, and the fission gas trapped in the fuel plates could cause considerable swelling. It was feared that some plates would eventually rupture.

Initial evidence of fuel growth was observed upon completion of the 85-Mwt power level phase of the program. Cursory examination of the plate assemblies (under the core water) revealed that the side plates were slightly distorted with some loss of the scale. There were no detectable physical defects in the fuel plates.

Scale spalling from the fuel elements presented some radiation problems, particularly in the subreactor room. The settling and accumulation of this scale in control rod rack housings and flanged appendages for the original forced-circulation nozzles created radiation levels up to 20 r/hr at contact. Blow down of these nozzles (to a dump tank) effected a reduction in surface radiation to 3-5 r/hr.

At power levels above 64 Mw, at which water carryover with the steam was encountered, fine particles of scale were entrained with the water. These particles did not plate out on any surfaces external to the reactor vessel, but were eventually collected in the full-flow condensate filters. Radiation levels external to the filter vessels normally are 10-15 mr/hr, but with carryover the levels increased to 20 r/hr. Irradiated particles of scale were also collected in the reactor water purification filters and resin beds. These vessels are shielded with lead (4 in. thick); therefore, no significant radiation levels were encountered.

Recognition and Evolution of the Scaling Problem

Perhaps the earliest recognition by Argonne of a potential problem with the aluminum dummy fuel elements, was Soppet's observation that they could be removed and replaced if corrosion is excessive. However, there was no specific inspection plan or alternate design that would effect the expedient replacement of the aluminum dummies.

The corrosion problem of the aluminum dummies surfaced during the first year of reactor operation. The material in this section is largely from a compendium of work by Breden, Charak and Leyse (1960). Other significant material is from a collection of EBWR Test Reports by Kolba (1960) These reports cover the phase 1 period beginning with plant startup during December, 1956 through the shutdown for conversion to 100 mW capacity that began during July 1959.I

During late 1956, hydrostatic testing of the pressure vessel gasket was carried out by using tap water in the system because sufficient quantities of pure water were not available. The aluminum-nickel dummy fuel M-388 dummy fuel elements were in the reactor vessel, but not the uranium fuel elements. The tap water remained in the system for two days. On reopening the vessel, extensive rusting of the vessel cladding was observed as a loose, reddish oxide film. Rather extensive corrosion of the aluminum-nickel dummy fuel elements was also observed. This corrosion was characterized by some blistering and was aggravated by galvanic attack near rivets holding stainless steel and aluminum-nickel in contact. A spectrographic analysis of the corrosion product revealed predominantly aluminum with trace amounts of iron, magnesium, nickel and silicon.

The reactor was started up during late 1956 and it operated intermittently until a shutdown for fuel and plant examinations during July, 1957. Two aluminum dummy elements and one fuel element were removed for inspection. Loose deposits were studied and found "...nonadherent and readily removed by a single wipe of the sponge." Spectrographic analyses revealed that , iron, nickel and aluminum were the major components.

The presence of a substantial amount of scale on the fuel elements was first noted during the examination of the core during January, 1958 after about one year of operation. Scale flaked off of one element as it was lifted from the core for underwater examination. Large flakes of scale rested on top of the core. Thickness of the scale (0.008 cm) was measured with a micrometer. Spectrographic analysis showed 67% aluminum, 25% nickel, and 8 %iron. The x-ray diffraction pattern was identical to samples from the corrosion layer on the aluminum dummies. Surprisingly, there was no apparent alarm with these discoveries. One investigator, (Kolba, 1960, Test Report No. 25 B) reported the above scale thickness as ranging from 0.008 to 0.0013 cm. However, he concluded, "Investigation during the first annual inspection of the EBWR core revealed no adverse or detrimental conditions which would necessitate further investigation to give complete confidence in future performance."

The EBWR operated at 61 Mw for several hours during March, 1958. After this run, the control rod bushing housings at the bottom of the reactor vessel were found to have very high levels of radioactivity (up to 60 r/hr on contact). The investigator (Kolba, 1960, Report No. 56) describes procedures for removing the material and identifies this as "...scale that flaked off the fuel elements during the 62 Mw operation and collected in pockets at the bottom of the reactor vessel." The major radioactive constituent was cobalt-58 and a spectrographic analysis was almost identical to scale deposits on the fuel elements.

The crud accumulating in the control rod bushings not only resulted in a buildup of activity in this area, but, in addition, caused mechanical interference with the movement of the rod. On one occasion it prevented the rod from going to is full "in" position by about 5 cm and made necessary some changes in the design of the bushing. (Breden, 1960)

Another overall fuel inspection was performed during April,1959. A gage block that was used to check the width of the flow channels became lodged in element ET-51. This element had operated in a relatively high flux location since startup and was removed for destructive examination. During examination in the Argonne hot cell a considerable amount of scale flaked off. See Fig. 3.

Fig. 3 Scale on plate from element ET-51. This photograph is 80% of actual width. The lighter regions have scale that has remained in place. The darker region near the center has lost its scale. Note the peeling between regions.

The thickness of this scale was about 0.013 cm. Density was 2.5 gm/cm^3 based on weight and volume (thickness x area). The scale was attracted by a magnet. Composition based on wet chemical, spectrographic and X-ray diffraction measurements yielded the following: boehmite, 80.6 %; nickel oxide, 12.6 %; iron oxide, 5.1%; silicon dioxide 1.6%.

The thermal conductivity of the scale was measured in apparatus specifically built to accommodate the EBWR scale (Breden, 1960). The value thus determined was in the range of 0.008 W/(cm^2)(C) and this is about one-half of the value cited by Wimuc in his summary above. At 50 W/cm^2, (the design peak heat flux at a power level of 20 mW) the temperature drop across 0.013 cm of scale thickness is about 80 C. However, the reactor operated for several hours at over 60 Mw during March 1958. That would yield fuel plate surface temperature of about 240 C in excess of the nominal design value of 255 C. However, with the descaling that was took place during the 60 Mw run, the high temperatures would not have been sustained. Of course, the maximum scale thicknesses that built up at the lower power levels could have exceeded 0.013 cm by a substantial margin. At a thickness of 0.020 cm, the scale temperature drop would be in the range of 370 C.

The phase 2 operations, 1960 through 1962, continued to be plagued by the scale that was deposited on the fuel elements during phase 1. The aluminum dummy elements were removed and replaced with additional rod-type elements, but flake-off of scale from the original elements significantly restrained plant operations. The following experience is documented in a second compilation of reports by Kolba (1964).

The reactor power level never exceeded 7 mW during 1960 and 1961, and the longest time at power was a run at 5 mW for 14 days during late September to early October 1961. There were about 140 days of power operation during 1962, 20 mW was reached during March, 40 Mw during July, and 60 mW for a 4 day run during September. The unit operated for nearly 30 days at 60 mW during it's final run that ended during early December, 1962. During this final run there were 8 brief operations above 60 mW, these reached 90, 75, 100, 80, 95, 75, 85 and 80 mW in that order. Following the final reactor run, very limited investigations classified the scale found in reactor components as primarily from reactor operation during phase 1 when the aluminum dummies were in place. This was evidenced by the high aluminum content.

One investigator's report included the following: "All indications point to a very dirty reactor when due to high water velocities and high temperature of the fuel plates, crud or scale flaked off the fuel plates. Not only was the blowdown tank loaded with crud and scale, but the full flow filters were highly radioactive, and the assemblies showed areas where the scale had. flaked off."

Indeed, the fuel plates had reached high temperatures in the scaled regions. The slotted side plates had some distortion as reported by Wimunc. The design peak heat flux for the phase 2 core at 100 mW is about 110 W/cm^2 and that is less than the design peak heat flux of 150 W/cm^2 at 61 mW during phase 1. That is because the nuclear design of the phase 2 core achieved a "flatter" power distribution. However, it is likely that fuel plate temperatures during phase 2 exceeded the fuel plate temperature reached during the 61 Mw run of phase 1. This is apparent because the side plates of some fuel elements became distorted during phase 2, while no such distortion occurred during phase 1.

There were no fuel plate failures even at operating temperatures well beyond design values. However, the total operating time at power that was achieved was very much less than expected when the EBWR project was launched. The aluminum dummies thus cost a fortune.

IMPACT OF SCALE ON NUCLEAR PLANT SAFETY

The EBWR scale significantly increased the fuel element operating temperatures, led to distortion of the fuel element side plates, and at times impacted the operation of the control rods. The point of the EBWR discussion below is that the adverse impact of the aluminum dummies was not covered in the hazards evaluation reports. This lack of timely dissemination of the adverse impact of aluminum components was unfortunate. This aspect is covered in the second item below, the Argonne Low Power Reactor (ALPR). The final item refers to present day nuclear power reactors in which the performance of the nuclear fuel elements has been highly satisfactory but is under close surveillance as oxidation of Zircaloy cladding may be approaching the regulatory limit of 17% wall thickness.

The EBWR

The initial Hazards Evaluation Report by West et al (undated) was written during 1955. It described the role of steam formation in limiting power excursion transients. There was no anticipation of scale formation in this report and an overall heat transfer coefficient from the fuel centerline to the boiling surface was reported as 1.07 W/(cm^2)(C). There was no correction of this heat transfer coefficient to account for scale formation in either of two subsequent revisions to the Hazards Summary by Wimunc and Harrer (1959 and 1960).

The December 1959 revision acknowledged that, *"Concurrent studies are in progress to determine the effect of scale deposits on the heat transfer surface of the fuel elements."* However, the brief paragraph erroneously concluded, *"Calculations to date indicate the temperature will not exceed 260 C."* The October 1960 version had the same paragraph, however, the paragraph now concluded that, *"Calculations to date indicate the temperatures may exceed 600 C."* Regarading scale, the only other revision was a sentence in the preface, *"From recent tests it appears that the scale formed on the Core 1 fuel plates may increase the fuel temperatures and the possibility of one or more elements failing at elevated powers cannot be ruled out."*

It is also worth noting that all of the versions of Hazards Evaluation Reports were withheld from general distribution until February 1962, following review by the U. S. Atomic Energy Commission. Even then, the revised hazards summary reports did not yield in-depth studies that reflected operating experience and the reports did not reference such.

The Argonne Low Power Reactor (ALPR)

The entire core of this boiling water reactor consisted of aluminum alloy structures with the exception of minor fittings of stainless steel. After about 30 months of operation and the expenditure of 40% of core life, the reactor was accidentally destroyed during preparations for startup following a shutdown for maintenance. The author believes that aluminum corrosion and scale formation was a significant factor in this accident.

The ALPR core. Like EBWR, the ALPR was a boiling water reactor. The power level was only 3 mW, its design mission was for remote military applications in which it would provide 300 kW of electricity and 400 kW of heat. The core consisted of a cylindrical arrangement of 40 square fuel elements with control rods and support structures.

The entire core of ALPR was fabricated from aluminum alloy (x-8001). This included the control rod channels as well as a set of boxes that each bounded four fuel elements. Five control rods were used for the 40 element core arrangement; one control rod was in the center of the core.

The fuel elements were plate type with nine plates per element, 88 cm long by 9.8 cm square. Each fuel plate was 3.05 mm thick, consisting of a 1.27 mm thick uranium-aluminum-nickel core with 0.89 mm thick aluminum alloy cladding. (Note the thickness of the cladding, this is pertinent to forthcoming discussions.) The five control blades had cadmium cores non-bonded to aluminum alloy cladding.

At the reactor design power of 3mW, the peak heat flux on a fuel plate was 205 kW/m^2 and the average was 71 kW/m^2. The operating pressure was 21 atm and the boiling temperature was 215 C. (For the phase 1 EBWR, the design peak heat flux at 20mW was 488 kW/m^2, pressure was 41 atm and the boiling temperature was 254 C.)

The ALPR destructive accident. The ALPR core was destroyed on January 3,1961 during preparations for restart. Investigations revealed that a plant operator was following a procedure that required manual withdrawal of the central control rod by about 5 cm in order to engage the rod to its mechanical drive. Perhaps the operator withdrew the control rod excessively, leading to the destructive pulse of reactor power. However, the reactor core was in a degraded condition prior to the accident and other factors likely contributed significantly to the disaster. The possibility that the control rod had been mechanically bound (stuck) was considered. The speculation was that the operator, in an attempt to free it, exerted a large upward force, and upon the sudden release of the rod inadvertently pulled it too far.

In testimony to an investigating board Zinn (1962) presented evidence and reasoning that refuted this speculation. It is not the intention of this paper to refute Zinn's remarks. However, it appears that Zinn may not have been informed of several pertinent factors. (At the time of the accident, Zinn was employed by a contractor that had assumed operation of the ALPR and had renamed it SL-1 (Stationary Low Power Reactor No. 1).

The unsatisfactory performance of aluminum alloy X-8001 was well recognized in the EBWR operations. The report by Breden et al of October, 1960 discussed the problems with scale formation and the likely impact on fuel element temperaures. It also described problems arising from descaling during plant operations including one instance of control rod binding, however, there was not a widespread discussion of the situation. Indeed, as noted above in the EBWR case, it was not until February 1962, over one year following the ALPR accident, that all versions of the EBWR Hazards Summary Report were released for general distribution.

Zinn opened his testimony by citing Report IDO-19311 (Final Report of the SL-1 Recovery Operation, General Electric, July 27, 1962). However, there is pertinent data in a subsequent report by General Electric, (IDO-19313, Additional Analysis of the SL-1 Excursion, November 21, 1962). This later report describes the examination of an unirradiated SL-1 control rod that had been used in an underwater mockup of rod withdrawal rates. Pertinent features that clarify the following discussion are in Fig. 4. The investigators observed: "The blade, upon sectioning, was found to be full of water. Considerable corrosion attack was evident in the cadmium and on the aluminum during cladding. Only one of the spot welds was holding."

Several sections from damaged control rod blades were examined with ingenious apparatus in the site radioactive materials laboratory (RML): "The blades examined in the RML were also noted to have few spot welds holding. It was first thought this had been caused by the incident, but the indication is that many of the spot welds may have been very weak to start with. .... Many of the edge welds of the post-incident blades were either cracked or split open. .... The existence of water in between the aluminum clad material would contribute to the corrosion observed on the cadmium blades and the clad material."

Tensile tests were cut from the cadmium sheet but not from the aluminum cladding. "The (cadmium) specimens were noted to have a fairly heavy corrosion coating which flake off during the tensile test."

Post-incident examinations of fuel plates included this note: "Examination of clad corrosion showed the corrosion to be fairly uniform at 0.06 to 0.09 mm thick." The report has no analysis to suggest that this corrosion was formed during the incident. Very likely, there was substantially greater fuel plate corrosion in the course of the 30 months of plant operation and the deposits periodically flaked off. It is also likely that the incident itself as well as the post-incident dissections led to scale flake-off.

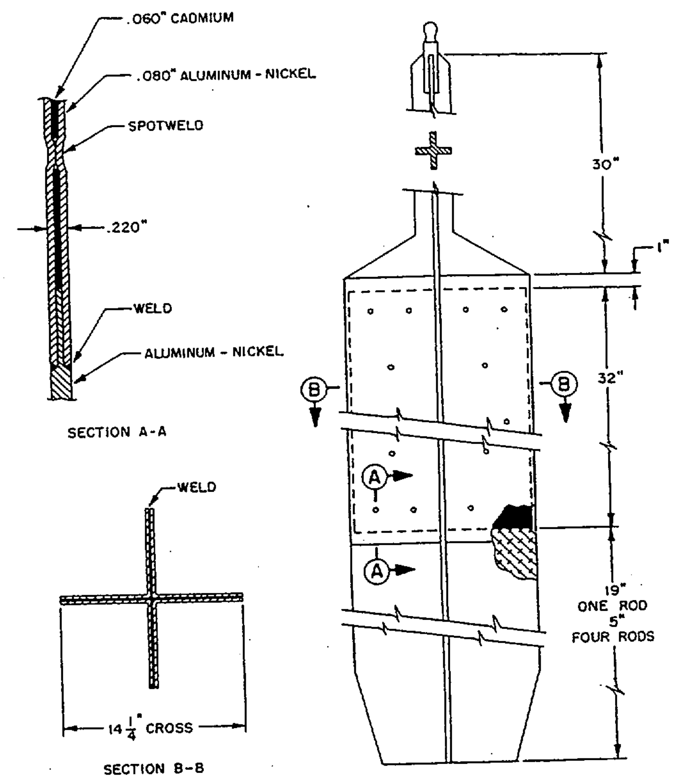

Fig. 4 Aluminum clad control rod. Note the spot welds.

The theoretical analysis of the ALPR accident. This analysis is not repeated here, but the impact of aluminum oxide corrosion deposits on the fuel plates in retarding the transient heat transfer to the water in the core was completely overlooked by the analysts. Indeed, that section is titled, "Steam Formation from Plates (smooth surfaces)." The steam formation is one of the turnaround mechanisms as water (moderator) is expelled from the core.

IDO-19313 cited experimental work at a university that determined the temperature overshoot before boiling commences under fast transient conditions. The specific relevance to ALPR is unclear as the analyst refers to excursions originating near the saturation temperature while ALPR was well below the saturation temperature.

Quoting from IDO-19313: "The thickness of the cladding of the SL-1 fuel elements had an important effect on the magnitude of the excursion. Because of the extremely short period, this 0.89 mm cladding became an effective thermal insulator and impeded the flow of heat to the reactor water where it could initiate shutdown of the reactor. A thinner cladding would have greatly reduced the maximum power level. Thus the effect of cladding thickness should be considered in the design of the fuel elements and in the estimation of the hazards of operation.

Unfortunately, a quantitative estimation of the cladding thickness and the magnitude of the excursion has not been completed at this time." Again, the impact of scale was ignored.

The thermal conductivity of aluminum is about 200 times greater than the conductivity of the corrosion on the fuel plate, A corrosion deposit only 0.00445 mm thick would have the same temperature gradient as 0.89 mm of aluminum cladding. Alternatively, the corrosion product thickness of 0.09 mm has 20 times the temperature gradient of the 0.89 mm aluminum cladding. Ignoring scale thus yields a grossly erroneous and incomplete analysis.

The redesigned ALPR. Another ALPR was never built. In his concluding remarks to the investigating board, Zinn stated that his employer, "... had recommended to the Commission, and the Commission approved, the design and construction of a new core which we believe would have improved the reactor by calling for adequate shutdown with any one rod removed, ... and by replacing aluminum with stainless steel a structural material." The proposed replacement of aluminum was certainly a sound design change. However, Zinn did not address the problem of scale formation arising from the aluminum cladding on the fuel elements.

Present Day Light Water Nuclear Power Plants

The power level of today's large nuclear power plants is in the range of 3000 mW. A typical example is Diablo Canyon Unit 1, a pressurized water reactor that is licensed to operate at about 3300 mW. The operating pressure is in the range of 170 atm and the water temperature is about 300 C. The reactor core is composed of an array of 193 fuel assemblies, each containing 264 fuel rods. These rods are composed of uranium dioxide pellets enclosed in zirconium alloy tubes with welded end plugs. The zirconium tubes are about 3.6 meters long with an outer diameter of 9.5 mm and a wall thickness in the range of 0.5mm. The fuel rods are supported in assemblies by a set of spring grid structures. Reactor control and shutdown functions are performed by the rod cluster control assemblies (RCCAs). The RCCAs are stainless steel tubes containing a silver-indium-cadmium absorber. These are located within zirconium guide tubes that are arranged among the fuel rods within the spring grid structures.

Corrosion of zircaloy cladding as well as deposition of corrosion products is now recognized as a key issue. There is an almost wild combination of nuclear power plant operators, nuclear fuel suppliers, federal regulators, universities, national laboratories, assorted consultants, lobbyists and non-profit organizations that are involved in the activites related to fuel element performance. There are some specious arguments as to whether certain situations are "merely" operational rather than safety problems, but there is no ignoring or lack of awareness of situations that were somewhat ignored in the past. The total situation is very complex and the complexity is amplified by the combination of operational realities, licensing rules and somewhat inapplicable physical modeling and related computer codes. The following is based on testimony to the Reactor Fuels Subcommittee of the Advisory Committee on Reactor Safeguards (ACRS) of the United States Nuclear Regulatory Commission, April 23 and 24, 1998. The items in this section are not covered in detail, but they are provided to illustrate the high degree of attention that is now directed to corrosion and crud deposition.

There are two basic factors: The first is that zircaloy cladding becomes oxidized to a depth that on occasion exceeds 0.1 mm. The other is that crud deposits form on the fuel rods, often preferentially in the region of the spring grid structures. Measurements of these factors have been produced, however new equipment needs to be invented and procedures implemented to precisely classify the thicknesses and distributions of these heat transfer barriers.

Specific data on zircaloy oxidation is not generally available. However, the following was disclosed at the ACRS meeting relative to a Spanish nuclear power plant: "... they also ended up observing some spalling and they also were having trouble correlating the data with the models."

The transcript also discloses the "... incomplete rod insertion issue. The safety concerns would be shutdown margin and potential precursor. Actually the issues turned out to be, in the final analysis, thimble tube growth and deformation such that the tubes were deformed and they slowed down and stopped to the rods." The transcript does not disclose the cause of the thimble tube deformations or the corrective actions. (Perhaps the corrosion and crud on the fuel rods effected a sufficient heat transfer resistance to yield growth and bending of the fuel such that adjacent control rod guide tubes were deformed.)

Another costly consequence of crud deposition, the axial offset anomaly (AOA), was discussed as follows: "This is a problem that has been around for a few years, but it has been around at kind of a low level. Various plants experienced axial offset anomaly, maybe 3 to 5 percent. It was somewhat of an annoyance. There were a few that got down to maybe 6 percent, and then last summer there was one plant that had a real problem, and basically they were beyond 15 percent. This really came to be a shutdown margin problem. How they handled it was continued operation. They continued to operate within their tech specs but they did operate for about four months at 70 percent power. As they got further into the cycle they were able to raise power slightly and at the beginning of this month, when they shut down, they were up to 86 percent power. Obviously it was a very big problem economically. The problem is the crud buildup. Crud buildup high in the core traps boron and pushes the flux to the bottom of the core." Investigation revealed that subcooled boiling in the upper part of the core was a factor. Corrective action included reducing the heat flux in those regions.

RIA and LOCA are design basis accidents (plant licensing issues) that are impacted by fuel element scale and crud deposition. Briefly, RIA means Reactivity Insertion Accident such as the ALPR accident. When the fuel rods are rapidly heated, the presence of scale delays heat transfer to the moderator (water) and this delays turnaround. LOCA means Loss Of Coolant Accident. If the reactor has a large line break, coolant is lost and the fuel rods heat up before emergency cooling water is injected. In the presence of pre-existing corrosion, the zircaloy fuel rods have a lessened margin to reaching an oxide thickness equal to 17 % of the zircaloy thickness. The 17 % number is a licensing restriction beyond which the rules dictate that fuel cladding may shatter during reflood quenching and thus block effective emergency cooling and initiate a meltdown.

A multi-million dollar test program has been initiated at the Argonne National Laboratory, and proceeds at a slow pace with deadlines that have now been extended to about 2005. The test plan (Argonne, 1998) consists of tensile and corrosion testing of sections of fuel rods that have had a significant operating history at selected power reactors. The program is thus designed to collect data from corroded (and perhaps crudded) fuel rods. Another multi-million dollar activity at a university is called Rod Bundle Heat Transfer and deals with LOCA heat transfer, however, there is no plan to include the impact of fuel rod corrosion. The report describes the relationship between the electrically heated fuel rod simulators and nuclear fuel rods. However, the reference nuclear fuel is as-built and not as-operated after several months of service. (Hochreiter, 2001).

Operational issues, safety issues, and crud. In the above cited ACRS meeting, a nuclear industry representative admonished the ACRS members as follows: "And as long as we stay within the tech spec, the operational limit, there shouldn't be any safety concern. I think experience has shown that has been the case. So, you know, it is really -- it is great to be on top of things. But some of those issues like AOA, they are really not safety issues, they are operational issues." Of course, it would have been appropriate for the ACRS to respond that their scope is not bounded by "tech specs." Furthermore, there is the ever present likelihood that analysis of operating experience may reveal inadequacies in the "tech specs."

Clearly, the experiences of decades ago, and more importantly the current operating problems, reveal that corrosion of fuel elements and crud deposition significantly alter the as-built condition of the nuclear power plants. There is an urgent need for improved inspection apparatus and the results of inspections should be promptly disclosed.

CONCLUSIONS

Early experience in the development of nuclear power plants yielded some tough lessons on the impact of fuel element scale on plant operating capability and plant safety. At EBWR, the fuel element scale severely restricted the development test program. At the ALPR, the severe corrosion of components may have been the root cause of the destructive accident. There was not a timely recognition of the factors that led to scale formation. In the case of EBWR, one designer recognized the potential lack of adequate corrosion resistance in the boiling water environment. In the case of the ALPR, there was no such recognition. Although the EBWR and ALPR both operated in the late 1950's and early 1960's there was apparently no open discussion of operating experience between the managers and the staffs of each unit.

After decades of operating experience, the formation and deposition of crud in current operating power reactors continue to pose new challenges. There is a lot of attention to the operating experience and there is substantial communication among the several stake holders. Inspections for corrosion thickness and characteriszation of crud are difficult, and further invention of equipment and techniques is necessary. Finally, among those who are employed to deal with the technology, there is no need classify scale and crud as either an operating nuisance or as a legal safety problem. The important activity is to collect and report the inspection data and analyze the related on-line operating experience for relevance to plant safety.

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PROPOSED ADDENDUM

Encouraged by the Conference Chair, Professor Hans Muller-Steinhagen, the participants engaged in "...free and extensive discussions..." during the lectures as well as the evening/night sessions in the bar and elsewhere.

This author, Leyse, was pointedly corrected when he asserted that no large power reactor would ever have scale formation within a fuel element comparable to the extensive scale deposits that have occurred during long-range (decades) of operation of the steam generators in pressurized water reactors. It was disclosed that experience at the River Bend boiling water reactor in the U. S. A. proved otherwise. At that plant an operational incident led to scale deposits that were apparently sufficient to bridge the channels between the fuel rods in certain locations.

Thus inspired, (upon his return to the United States) the author reviewed a report of the River Bend experience that was voluntarily submitted to the U. S. Nuclear Regulatory Commission (Licensee Event Report 50-458/99-016-00). The report refers to an unusually heavy deposition of crud on the fuel bundles and related fuel rod "perforations" (failures). Here are some direct quotes: "The rods with perforations had heavy crud with clumpy formations. Limited spalling patterns were observed on the highest power rods."

The report does not disclose the number of fuel rod failures, but it does disclose that all of the failures were confined to a "new" set of seven fuel bundles with the designation HGE. The report emphasizes that, "Even with the fuel cladding defects experienced during Cycle 8, the plant continued to operate within the bounds of its Operating License, including the Technical Specifications, and its licensing basis, including the Updated Safety Analysis Report (USAR). These limitations provide defense-in-depth for the public health and safety. Fuel cladding failure is not an unanticipated condition, but rather is an integral part of the licensing basis of the River Bend Station. Fuel cladding defects are acceptable to the extent that they do not jeopardize radiation protection limits established in the plant Technical Specifications and other licensing basis documents."

However, the report is generally qualitative, not quantitative. The composition of the crud is merely described as iron with copper in layers adjacent to the fuel rod cladding. There is no reported analysis of the fuel rod cladding temperatures that were reached during various phases of the Cycle 8.

In the spirit of free and open discussion that this conference encourages, the author believes that the River Bend Station did not comply with loss-of-coolant-accident (LOCA) or Reactivity Insertion Accident (RIA) limitations during its Cycle 8. It appears that the Technical Specifications and other licensing basis documents are either deficient or not enforced or both. Continuing plant operation as fuel failures multiply is unwise.

Entergy

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Rick J. King
Director
Nuclear Safety Assurance

March 1, 2000

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: River Bend Station - Unit 1
Docket No. 50-458
License No. NPF-47
Licensee Event Report 50-458/99-016-00

File Nos. G9.5, G9.25.1.3

RBG-45275
RBF1-00-0030

Ladies and Gentlemen:

Enclosed is the subject Licensee Event Report. The report is being filed voluntarily, due to the potential generic applicability of this condition. No commitments are identified in this report.

Sincerely,

Rick J King

RJK/dhw
Attachment
Enclosure

cc: U. S. Nuclear Regulatory Commission
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NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(6-1998)

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001

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TITLE (4)
Thermally-Induced Accelerated Corrosion of BWR Fuel

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIA NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
04	20	1999	1999	--016--	00	03	01	2000	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	5	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(viii)
POWER LEVEL (10)	0		20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)	X	OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
D. N. Lorfing, Supervisor – Licensing	225-381-4157

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On April 20, 1999, with the plant in Mode 5 for a refueling outage, plant personnel documented an unusually hea deposition of crud on fuel bundles (*AC*) removed following the preceding operating cycle (Cycle 8). A root cau investigation was performed. The information gathered and conclusions reached during the root cause process are of such relevance to the industry and the NRC that a voluntary report was deemed appropriate.

An exact root cause was not identified, but the investigation indicates that multiple factors contributed to an accelerated corrosion of the fuel cladding in the highest-powered fuel bundles during Cycle 8. The heaviest deposition was discovered on the first-cycle fuel. Corrective actions were developed through River Bend's root cause analysis process, and these will aid in preventing recurrence of the crud deposition which induced the corrosion by thermally insulating the fuel rods.

Safety significance was evaluated for the increased crud level and for the clad perforations. The significance of perforations was low, since they are considered in the licensing basis. Significance of the elevated crud level wa determined to be acceptable through a process which included engineering judgement, combined with analyses various plant conditions.

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REPORTED CONDITION

On April 20, 1999, with the plant in Mode 5 for a refueling outage, plant personnel documented an unusually heavy deposition of crud on fuel bundles (*AC*) removed following the preceding operating cycle (Cycle 8). (Crud is a colloquial term for corrosion and wear products, e.g., rust particles, that become activated when exposed to radiation.) A root cause investigation did not reveal that the higher-than-normal crud levels existinc at River Bend Station (RBS) warranted a report pursuant to 10 CFR 50.72 or 10 CFR 50.73. The information gathered and conclusions reached during the root cause process, however, are of sufficient relevance to the industry and the NRC that a voluntary report was deemed appropriate. Therefore, Entergy Operations, Inc. (EOI), is submitting a voluntary event report to document the thermally induced accelerated corrosion phenomenon discovered at RBS.

BACKGROUND

On September 18, 1998, a fuel element cladding defect was indicated by offgas (*WF*) chemistry sample data Operations personnel requested the sample after noting an offgas pretreatment alarm (*RA*) during control roc drive (*AA*) operability testing. Immediate actions included re-sampling to verify results, informing plant management, and increasing the sampling frequency to once per day. Actions were taken in accordance with procedure ADM-0084, "Fuel Integrity Monitoring Program and Failed Fuel Action Plan." Operations personnel also verified that the thermal limits remained within the plant Technical Specifications 3.2.1, 3.2.2, and 3.3.3. A report was issued, pursuant to 10 CFR 50.72(b)(2)(vi), when the State of Louisiana was notified of the indicatic

Additional fuel element cladding defects were indicated during the remainder of the operating cycle. These additional fuel element cladding defects were indicated by increases in the offgas activity and the guidance of ADM-0084 was followed. Reports were issued, pursuant to 10 CFR 50.72(b)(2)(vi), when the State of Louisiar was notified of the indications. Reactor power in the vicinity of the indicated defects was suppressed through control rod (*AC*) insertion, and this successfully mitigated the activity release consequences of the defects. Power operation continued until April 3, 1999, when RBS shutdown for refueling outage no. 8 (RF-8).

The bundles suspected to have experienced fuel clad perforations were those first-cycle bundles loaded into th reactor core for the previous Cycle 8 operation. These first cycle bundles were manufactured with a serial number which included the designation HGE. Visual inspection and telescopic sipping of the bundles during the refueling outage confirmed that all of the perforations did occur in a total of seven HGE fuel bundles.

Upon initial visual examination of selected fuel bundles with potential fuel cladding defects, the fuel inspectors noticed an unusually heavy deposition of crud on the fuel pins. Following the identification of the crud buildup, multidiscipline team was instituted to determine the relationship of this material to the fuel element cladding defects. Additional fuel bundles were selected for examination, and other actions were initiated to address the issues.

INVESTIGATION

Fuel inspection was conducted at RBS during RF-8 to determine the cause and extent of the fuel cladding defects, and to determine the population of fuel bundles acceptable for use in the next cycle. Inspections included not only HGE fuel (i.e., first-burned fuel), but also GGE (twice burned) and YJ8 (thrice burned) fuel in the reactor (*RCT*) during Cycle 8. Bundles that were not operated in the core during this cycle were inspecte to establish a baseline for the observations. Bundles from Cycle 6 and Cycle 7 at RBS were inspected. Inspection data were also obtained for bundles that operated in similar plants that have operated with high feedwater iron concentrations.

The following are observations specific to the HGE bundles, which were the only bundles that experienced cladding perforations.

- The perforations were due to cladding corrosion, which appears to be related to the thermal effects of high crud loading. Limited spalling patterns were observed on the highest power rods.
- The rods with perforations had heavy crud with clumpy formations.
- The perforations were at about the 50" elevation on the rods.
- The perforations were in HGE (first-burned) fuel.
- The affected HGE bundles had Linear Heat Generation Rates (LHGR) at the 50" level that were in the top 3% of the entire core power levels during the first control rod sequence of Cycle 8 operation.
- All but one of the affected bundles had a shallow control blade adjacent to the bundle during the first contrc rod sequence.
- The bundles with perforations were in the high-powered core ring.

In determining causal factors for the observations noted above, various facets were investigated. The investigation is divided into two sections: an investigation of the crud itself; and an investigation of the differences in operational parameters between Cycle 7, which had no clad defects, and Cycle 8, which had multiple clad defects.

<u>Crud</u>

The amount of crud observed during the fuel inspections was higher than normal. The observed iron deposits are the result of the input from the feedwater stream combined with a chemistry excursion which occurred duri startup from RF-7. The chemistry excursion manifested itself as a conductivity excursion that began at the poi of heater drain (*SM*) pumped-forward operation and persisted for approximately three weeks (10/23/97 to 11/15/97). The conductivity excursion, which qualitatively accounts for the balance of the iron noted on the fue beyond that accounted for in the feedwater stream, is believed to have contributed to the onset of the cladding corrosion condition. At the time of the excursion, there was no reason to suspect it would affect crud depositio on the fuel.

In response to this condition, the investigation included an examination of locations that might contain an inventory of iron oxides available for future release. These areas included the main condenser (*SG*) and the condensate storage tank (*KA*) by direct visual and sampling, and the reactor vessel by running the reactor

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water cleanup (RWCU) (*CE*) loop aligned to the bottom head, where no flow restriction was noted. Only the condenser exhibited any significant inventory of iron oxides and copper, which was removed during RF-8. Flov accelerated corrosion (FAC) program results did not indicate unusual wear that could account for the level of iron found in the vessel.

Chemistry analysis history was reviewed for any significant anomalies that could have caused either the crud deposition, or the accelerated corrosion. The one extended period of a conductivity spike, with a gradual returi to normal over a three-week period early in the cycle, was unusual. The review indicates that plant parameters were within the EPRI guidelines for operation of the plant.

The potential for a chemical intrusion (as a direct corrosive agent) was also considered. Data for plant chemist during RF-7, including the residual heat removal (RHR) (*BO*) chemical cleaning conducted for the first time during the outage, and the forced outage in April 1998 were reviewed. No evidence of a significant chemical intrusion thought to be capable of affecting the core was identified.

Cycle Differences

A synergy among various parameters related to plant chemistry and core operation is required, in conjunction with the iron deposits, to adequately explain the corrosion phenomenon. A review of parameters that changed in any significant way between Cycle 7 and Cycle 8 was performed.

- The amount of iron input to the reactor vessel increased by ~70% in Cycle 8, versus Cycle 7, due in part to the removal of low cross-linked resins from service in the condensate demineralizers (*SF*). This removal was done because of sulfate bleed-through associated with this particular resin type. An iron oxide crud layer on the fuel provides a means to concentrate soluble elements such as copper.
- The amount of copper input to the reactor vessel increased by ~30% in Cycle 8 versus Cycle 7, again due t the removal of low cross-linked resins from service in the condensate demineralizers. An additional source of increasing copper is the "blinding" effect of higher iron on the demineralizers copper removal efficiency. Copper has been previously implicated as an agent of local cladding corrosion in the BWR fleet. Analysis c the crud layers indicated that copper had concentrated in the crud layer adjacent to the cladding.
- Zinc was injected into the feedwater system in significant quantities for the first time in Cycle 8. However, the amount of zinc injected and ultimately deposited on the fuel was unremarkable, as compared to the BW fleet experience. There is no known corrosion or corrosive agent concentration mechanism associated witł zinc injection. This is not believed to be a factor in the crud formation.
- The plant operated in the Maximum Extended Load Line Limit Analysis (MELLLA) domain for the first time following RF-7. While this allowed plant operation at lower overall core flows, the locations of the fuel failures were not the locations of lowest flow. The failure locations show a strong correlation to peak nodal powers (as expected for a duty-related failure mechanism such as corrosion), but do not show such a correlation to low bundle flow. The lower flows due to MELLLA would only be a minor aggravating factor fo crud deposition. Bundle inspections at other BWRs with high feedwater iron concentrations and MELLLA operation do not indicate any significant increases in crud levels due to MELLLA operation.

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ROOT CAUSE

Absent a single event or clear indication of a cause, it is concluded that an early cycle event, indicated by the prolonged early-in-cycle conductivity transient, combined with higher iron and copper levels, resulted in an unusual crud deposition that initiated the process which led to accelerated cladding localized corrosion-inducec perforations. None of the individual factors, alone, have caused the corrosion phenomenon at plants in the pas as evidenced by a review of operating experience.

The higher input of iron and copper during the operating cycle, with a chemistry excursion early in the operatin cycle, produced the unusual crud deposition and composition observed during the visual inspections. The concentration of copper in the crud layer provides an attack mechanism to foster the observed corrosion. It is significant to note that the crud deposition peaked at approximately the 50" level, which is where the primary clad perforations also occurred. The 50" level corresponds to the power peak for the first (A2) rod sequence in six of the seven perforation locations. The early-cycle conductivity increase occurred during the A2 rod sequence.

It is a well known relationship that Zircaloy corrosion increases with increasing clad temperature. It is not unexpected to find that the corrosion occurred in the highest-powered regions of the core. The formation of a Zircaloy oxide layer is dependent on temperature. As the crud loading on the fuel became heavier, it increasec thermal resistance and raised clad temperature, which resulted in increased clad oxidation. The presence of high copper concentrations under these conditions tends to aggravate the situation. Soluble copper will concentrate in the oxide layers adjacent to the fuel rod. Differences in copper oxide growth and Zircaloy oxide growth can result in a higher insulating effect. The increased oxidation thickness results in increased thermal resistance. This becomes an autocatalytic process, which proceeds until the combination of higher temperatur crud, and copper result in clad perforation.

This process resulted in perforation only for the highest-powered bundles (the HGE batch). Measured Zircaloy oxide thickness on high power unfailed HGE bundles was up to 6-mils at the 50" level, where the cladding perforations occurred. By contrast, the lower power GGE bundles (initially inserted for Cycle 7) experienced fu oxide layers of typically only 1 mil, which is in the normal range. This demonstrates that without power to drive the oxidation process, the crud deposition does not result in a higher thickness of Zircaloy oxide. The GGE bundles did not experience fuel perforations.

It is therefore concluded that the elevated crud and the corrosion were likely due to a combination of various plant chemistry and operating characteristics that changed substantially from Cycle 7 to Cycle 8. The corrosior mechanism is likely due to the presence of contributing agents (primarily copper) within the crud on the higher-powered bundles. Absent any of these factors, the corrosion would likely not have been experienced to the degree observed.

CORRECTIVE ACTIONS

The root cause analysis report for this condition identifies corrective actions being taken at River Bend Station to address the issues. These include immediate actions taken for the startup and operation of the reactor for Cycle 9, and long term actions to be completed throughout the operating cycle and the subsequent refueling outage. These actions are being tracked in the RBS corrective action program.

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LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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SAFETY EVALUATION

Effects of Fuel Cladding Defects

The safety significance of the fuel cladding defects that resulted in fuel failure is low. Continuous monitoring of the reactor coolant system offgas provides early indication of the problem, allowing time to take the appropriate actions to monitor and mitigate the activity release consequences of the perforations. The plant's licensing ba and safety analysis assumes that fuel cladding defects can occur during normal operation. Even with the fuel cladding defects experienced during Cycle 8, the plant continued to operate within the bounds of its Operating License, including the Technical Specifications, and its licensing basis, including the Updated Safety Analysis Report (USAR). Together, these documents contain NRC-approved limitations for operating parameters such as reactor coolant system activity, gaseous radioactive effluents, and occupational radiation exposure. These limitations provide defense-in-depth protection for the public health and safety. Fuel cladding failure is not an unanticipated condition, but rather is an integral part of the licensing basis of RBS. Fuel cladding defects are acceptable to the extent that they do not jeopardize radiation protection limits established in the plant Technica Specifications and other licensing basis documents.

Effects of Crud

The safety significance of the effect of the elevated crud on Cycle 8 operation was evaluated. The results, as summarized below, demonstrate, based on previously performed analyses and engineering judgment, that the safety significance of the elevated crud levels is acceptable.

- The Thermal-Mechanical evaluation is intended to provide protection to thermal mechanical limits, such as cladding strain. Increased crud on HGE would accelerate the cladding oxidation process. An assessment the number of "failed" fuel rods (based on exceeding LHGR limits derived from the thermal mechanical limi indicates that the dose consequences would represent only a small fraction of 10CFR100 limits, and therefore the River Bend Cycle 8 condition was of acceptable safety significance.
- Given the inherent conservatism in the Safety Limit Minimum Critical Power Ratio (SLMCPR) process and the fact that suppression rods were required during the Cycle 8 operation, it is concluded that the SLMCPR would remain valid for operation in Cycle 8 under the assumed elevated crud conditions.
- The evaluation of operational transients concluded that the Minimum Critical Power Ratio (MCPR) operatin limits that were established for Cycle 8 operation would not ensure that at least 99.9% of the rods in the co would avoid boiling transition for an abnormal operational occurrence. However, an assessment of the number of "failed" fuel rods indicates that the dose consequences would represent only a small fraction of 10CFR100 limits. Therefore, the River Bend Cycle 8 condition was of acceptable safety significance.
- The peak clad temperature (PCT) for HGE fuel was calculated to have been 1700°F or less. This still demonstrates substantial margin to the 10CFR50.46 PCT limit of 2200°F. Note that excluding the oxide buildup during steady state operation, the peak local clad oxidation due to LOCA would remain well below the 17% requirement of 10 CFR 50.46, as there would have been no appreciable change in the percent of clad participating in the Metal-Water Reaction under LOCA conditions.

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Other analyses such as nuclear reactivity, over-pressure protection, and stability remain unaffected by the elevated crud.

PREVIOUS OCCURRENCE EVALUATION

Previous Fuel Cladding Defects at RBS

Previous fuel clad defects and perforations at River Bend were reviewed. No previous occurrences were applicable to the RF-8 fuel conditions, since the previous occurrences did not exhibit the heavy crud and the thermally induced accelerated corrosion.

Related Defects (Caused by Corrosion) at Other Facilities

No previous occurrences were found at other facilities that were similar to th occurrence at RBS. In the NRC's Safety Evaluation Report (SER) (NUREG-0989) for RBS, external corrosion and crud buildup on the waterside the fuel was discussed. The NRC notes that in the late 1970s and early 1980s, certain of these types of perforations were referred to as "crud-induced local corrosion (CILC) failures." A contributor to CILC was an unusual composition of metallic crud. The NRC further notes that the corrosion was reportedly associated with variably high copper concentration in the core coolant water and a minor anomaly in the Zircaloy cladding metallurgy, although both the water chemistry and cladding metallurgy were within allowable specifications. Cr deposits, aside from the CILC phenomenon, were expected even with improvements in newer plants such as RBS. Unlike the classic CILC, and even though a crud layer existed with high copper concentration, corrosion levels were driven more by crud thickness rather than corrosion caused by local cladding conditions.

Note: The Energy Industry Identification System (EIIS) component/system number is indicated by a parenthes after the affected component/system. (Example: (*XX*))