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Notes on Idaho National Laboratory Advanced Test Reactor FOIA Documents
Documents Received from DOE/ID 12/30/08 (third installment shipment on original EDI/KYNF FOIA 4/10/08 request).

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*After a cursory review (with emphasis on more current documents) of all documents in this "third" individual shipment, the notes generated below reflect only the relevant information related to serious ATR safety issues. These notes are NOT listed in any order of importance. Notes in (**Bold**) are document titles; notes in ("quotes") are drawn from the cited document; notes in [brackets] are Broscious' editorial comments. Citations ("Response to Request No…") that often contained multiple individual documents, are DOE/ID's own reference to EDI/KYNF's original FOIA request 4/10/08. When Notes indicate DOE/ID **redaction**, they are in the form of "blacked-out" for a sentence/paragraph and complete" blank-out" for whole pages; and/or only select pages of longer documents. EDI comments on the first two FOIA shipments dated 7/15/08 and 8/21/08 are covered in separate EDI reports.*

1. ATR In-service Inspection Plan Fourth Inspection Interval February 2006 to January 2015; Doc. ID: PLN-859, 12/18/06. Response to Request No. 4.

2.4.3; Heat Exchangers; "Several of the heat exchangers are known to have flaws in the cladding, especially in the area of the tube-sheet-to-channel welds. Radiography has been the examination technique for the heat exchanger welds. Radiography requires the PCS (primary coolant system) to be drained and, thus, has been performed only during core internals change-out outages. However, examination has been limited in the previous two outages to the M-2 heat exchanger only. This practice was not in strict accordance per the requirements of [current Code] Section XI. Section XI requires that all of the heat exchangers be examined in the first inspection interval, but allows examination of only one in subsequent intervals." [pg. 40]

Appendix C: 1; "The ATR Primary Coolant System (PCS) and the original six loops at the ATR were designed and constructed in the early 1960's using the criteria of [American National Standards Institute] ANSI B31.1-1955. But re-analysis does not change the code of Record for the system unless all the requirements of the new code editions are met, (analysis, material qualifications, pre-service inspections, and so forth). Since this was not done, the code of record for the ATR PCS is ASA Code for Pressure Piping B31.1-1955."

"Again, for reason stated above, the Record for Loops 2E-NW, 2C-S, and 2C-E are [old original] ASA Code for Pressure Piping B31.1-1955." [pg 115]

2. Interoffice Memorandum, INL, March 29, 2005, Plant Systems Engineering Review for Facility Certification No.29, From D.J. Schooner. [Request No. 4c]

Core Internals Chang-out [CIC] VI; "The C/2 N-16 tube has historically failed two to four years following the CIC. The apparent design flaw with the C/2 N-16 tube has not been investigated and corrected so it can be expected to fail two to four years from now."

Seismic Fragilities; "Of the seismic issues that have been identified since then, Plant Systems Engineering and Nuclear Safety and Engineering have determined which of these upgrades are required prior to full power operation of the ATR. These issues are currently being worked and will be required to be completed prior to startup for the full power operation. Notable open issues include the outer area buildings such as TRA-608, TRA-609, TRA-619, TRA-671 and the deep-well pump hour buildings. In addition the ATR underground diesel day-tank (TRA-776) and fire protection piping (ARES 2004) have also been identified." [pg.3]

"The ATR PCS/SCS heat exchangers are operating beyond 200% of their 20-year design life. To date, the only failure has been a single case of pitting corrosion in the heat exchange shell of 670-M-85." [pg. 4]

"Building Confinement; Review of the recent annual building leakrate data indicate that the leakage was above the 125% acceptance line. In addition all of the primary dampers that are required to close during the leakrate showed signs of seal leakage. BDM-1-5A continues to fail to open in cold weather." [pg.4]

Diesel Power to Deep-well Pumps; "Deep-well # 3 had to be disconnected from the commercial feed and can only be run from the portable generator. In addition only having one deep-well on diesel power does not provide any redundancy." [pg.4]

Regulating Rod (Reg Rod); "During removal of the reg rods one of the followers detached and fell into the tank…due to heavy corrosion. The new reg rod followers, however, are chrome plated and can be expected to experience the same failure mechanism. The metallurgical evaluation suggests that within two to three years the reg rod followers should be replaced with a different metal such as zircaloy." [pg.5]

In Vessel Post Accident Monitoring System (IVPAMS); "Design life of the electrically conductive elements in the probe, to not more than ten years. Because it is likely that the IVPAMS in tank hardware will start to fail in the near future, replacement of the system should be considered." [pg. 6]

Emergency Firewater Injection System Vessel Level Reference Legs; [A]re showing signs of pitting corrosion. Other in tank aluminum structures that have never been inspected are the reflector support tank and the core reflector tank." [pg.6]

Electrical Distribution; "Although the electrical utility upgrade project updated a significant amount of the switchgear there is a fair amount of switchgear that is well beyond its design life.

This includes the 50 year-old switch gear in building 609 and the 40 year old E-3 switch gear in the ATR." [pg.6]

ATR 40 Ton Crane; [C]racks in welds on the trolley frame have already been observed and this weld evaluation demonstrates that many welds are highly stressed."

Raw Water Overhead Tank; "The tank…needs to be refurbished…. [but] the project has not been funded." [pg. 10]

Emergency Firewater Injection System; [L]eakage was noted on the bottom head EFSI supply line…..due to corrosion."

High Demineralized Water Heat Exchanger; "[W]as exhibiting signs of possible loose brackets or tubing …..and substantial buildup of corrosion products. [T]he heat exchanger should be replaced." [pg.10]

DRMS Computers; "The DRMS continues to have a high failure rate and many of the spare parts for this system are no longer available. This leaves only on operational computer for the DRMS system. [F]ailure of the remaining computer would cause operations to enter an Abnormal Operating Procedure which would require a significant amount of RCT manpower to adequately monitor the [ATR] plant." [pg. 10]

Spare Safety Rod Drive; "There is currently no spare safety rod drive. In addition there are two other new safety rod drives that have deficiencies that prevent them from being used." [pg.11]

Instrument Thimbles; "The thimble purge system HVE-23 has shown degraded flow through the thimbles….but could cause failure of the nuclear instruments." [pg.11]

Warm Waste Pumps and Motors; "The warm waste pumps and motors are aging. They are beyond their design life."

Upper Emergency Firewater Injection System Control Valves; "[H]ad significant amount of leakage from valve packing. However, the replacement valves were not built to American Society of Mechanical Engineers Section III code of construction.."

Warm Waste System Control System; "The TRA-605 warm waste treatment facility control system is no longer reliable." [pg.12]

Uninterruptable Power Supply; "The utility UPS still needs to be replaced and funding needs to be provided to perform this upgrade."

Reactor Data Acquisition System; [U]pgrade should be funded to allow the VAX computers to be removed and to incorporate the console display system."

Regulator Rod; "The reg rod drives were not included in this upgrade."

Evaporation Ponds; The design life of the liners is 20 years, which will be 2012. The 10-year Plan or lifetime extension program should reflect new liners prior to 2012."

M-10 Emergency Pump; "The safety posture of the ATR would be significantly increased if the M-10 emergency pump were to have a more reliable battery backed power supply." [pg.13]

TRA-609 Switchgear Failure to Operate; "There has been one occasion since he last FCR where the M-6 diesel failed to load automatically. Having the M-6 diesel operable is a SAR-153 commitment and therefore operations need to have controls in place to prevent the M-42, M-43 or cross-tie breakers from being racked out during reactor operations." [pg.13]

Molyetek Recorders; "The recorders are approaching the end of life."

3. Interoffice Memorandum, April 18, 2005, from J.F. Graham, [Response 4g].

Beryllium Waste Generation; "Currently there is no disposal facility that can receive the irradiated beryllium waste."

4. ATR Vessel Vent Valve Instillation, EG&G Facility Change Form, 1988; Reactor Vessel Vent System issued 6/22/93, Doc. No. 7.3.12.3.21; [Response to Request No. 9i];

This report, although dated, is HEAVILY redacted with current DOE/ID "Ex. 2, 3" notations (FOIA Exemptions 2 and 3 hand written in blue ink at each redaction); Redactions include 31 individual paragraphs plus 13 whole pages. See Mark Sullivan's legal argument on exemptions 2 & 3 to FOIA releases.

"The purpose of the reactor vessel vent system which could be generated by decay heat in the reactor vessel during a Complete Loss of Flow Accident (CLOFA)." [Pg. 1]

"Install two ATR vessel vent valves and associated instruments and controls per Functions and Design Requirements Document. This system will vent steam and water into the ATR warm waste system during a Complete Loss of Flow Accident (CLOFA). The vessel vent valves provide the capability to remotely vent and relieve this pressure, thus allowing the FIS to operate under these CLOFA conditions. In the worst case CLOFA conditions steam could vent into portions of the reactor building second basement. The Vessel Vent System (VVS) provides a depressurization vent for the ATR vessel, ensuring that the Emergency Firewater Injection System (EFIS) can maintain water over the ATR core following a complete loss of flow accident. "

Reactor Vessel Vent System (VVS); "The VVS provides a depressurization vent for the ATR vessel, insuring that the Emergency Firewater Injection System (EFIS) can maintain water over the ATR core following a complete loss of [coolant] flow accident (CLOFA)." [pg. vi]

Seismic Requirement; "A Richter magnitude 7 earthquake with a maximum horizontal ground acceleration of 0.24 g at TRA (Modified Mercalli intensity of 8) has been used to define the Safe Shutdown Earthquake (SSE) for ATR. The Operating Basis Earthquake (OBE) for ATR is defined to have a magnitude equal to one-half of the SSE (i.e., modified Mercalli intensity of 4). Supports shall be designed in accordance with [old outdated] ASME Section III Subsection NF." [pg. 8] [There is no apparent provision in current American National Standards Institute (ANSI) seismic standards that allows DOE to unilaterally cut in half the Safe Shutdown Earthquake (SSE) for ATR Operating Basis Earthquake (OBE).]

"Under CLOFA conditions, steam would be vented by the VVS to ATR Warm Waste System. A study was made (EDF PRPTS-ATR-224) to determine whether that system is capable of handling the amount of steam that would be vented. The study concluded that, in the worst case, steam would escape from the Warm Waste System through open drains in the reactor second basement area. This area is within the reactor confinement area, and such venting would be acceptable under accident conditions for the purpose of controlling discharge." [pg. 33]

"Another limitation of the VVS is that, if inadvertently actuated, it will depressurize the Primary Coolant System." [pg. 34]

"The Vessel Vent System must not be activated inadvertently. If the VVS is activated following a CLOFA, the ATR Warm Waste System has the capacity to handle the volume of steam as long as fan HVE-17A or -17B is operating. However, if these fans were to fail, it is probable that steam would back up out of loop vent collection trough drains, sample cabinet drains, transmitter cabinet drains, etc. This could result in the spread of airborne radioactivity. However, the vessel vent discharge would be controlled within the ATR containment, which in this case is acceptable in light of the low probability of the event and the consequences of not venting." [pg. 35]

"If the VVS is tested while the reactor is up, the loss of system pressure will cause a reactor scram." [pg. 38]

"Electrical power is required to energize the two solenoid valves in the Vessel Vent System. Loss of electrical power for an extended period is unlikely to occur." [pg. 40]

"If the associated vent valve piping downstream of the valves should break during a CLOFA, the nozzle trench would be flooded with steam and water, If the vent valve piping up stream of the valves should break during reactor operation, the nozzle trench would be flooded with water. For venting purposes, this is a fail-safe condition." [pg. 41]

"The Vessel Vent System is designed, constructed, and installed to Uniform Building Code Zone 3 requirements, as required by DOE-ID architectural Engineering Standards. The possibility that damage would be sustained as a result of earthquake is therefore considered remote. Recovery would be dependent upon the local severity. Seismic forces in the region of the VVS piping could cause the system to be inoperative. But a break in the piping for both vent valves is unlikely." [pg. 42]

"The Vessel Vent System is designed such that channel maintenance can be performed while the reactor is operating. The channels have a design life of 10 years plus. The only components in a high-radiation area are the two vent valves, each of which has "N" stamp and were designed for the exposure expected in this area." [pg. 43]

5. Interoffice Memorandum, April 18, 2005, From C.D. Morgan, Facility Certification Report (FCR), Outstanding Radcon Problems Revision 1, Idaho National Laboratory. Response to Request No. 9d.

"There is a plant upgrade that has been stopped, that would convert the RMS 4 instruments (CAM 8 and 9) from the GNC CAMSs to the Digital Radiation Monitoring System (DRMS). There are currently no spare parts for repairing the GNC CAMS. The upgrade needs to be completed and the tracer lab RAM and GNC CAM systems need removed from the ATR facility. The removal of the molytech recorders should also be included."

"The High Level Radiation Monitoring system is not working as intended. The local indications are disabled, and the chart recorders are out of service and not repairable."

"DRMS continues to have numerous faults with the software that may not be possible to be overcome. The DRMS was put in the 10-year plan for replacement back in 2002, but there is still no fast path to ensure that this can be maintained."

"The existing software for the DRMS continues to fail with communication and control parameters and the new software needs to be completed and installed. This will ensure that the system will be reliable until a replacement is procured."

"The existing hardware has frequent failures and repair is questionable with each hardware failure. There is no current vendor for hardware spare parts. A contract with a DRMS hardware provider needs to be completed."

"An Abnormal Operating Procedure has been issued which identifies the minimum monitoring equipment for personnel protection during ATR operations and corrective action to be taken to provide equivalent interim monitoring."

6. Recommendation for Upgrade of Radiation Monitoring at the Idaho National Laboratory Reactor Technology Complex, August 2007, ANN Inc., et al., Attached to Response to Request No. 9d .

[Only page 2-41 was attached, thus the rest was redacted]

6.a. Chapter 12 Radiological Protection Upgraded Final Safety Analysis Report for Advanced Test Reactor, SAR-153, 2/05/08, page 12-69, Rev. 09. Attached to above "Response to Request No. 9d."

[Only page 12-69 was attached, thus the rest was redacted]

6.b. Advanced Test Reactor, Retire High-level Radiation Monitoring System, TRA-USQ-2005-097, Unanswered Safety Question, USQ Process Proposed Change Form, 6/10/04. Attached to above "Response to Request No. 9d."

"12.5.2.3.3 High Level Radiation Monitoring System; The High-Level Radiation Monitoring System (HLRMS) was used to detect and measure radiation levels in the confinement area that would result from any reactor accident up to the levels expected from a maximum hypothetical accident. The HLRMS is out of service because it no longer has a functional basis."

6.c. Radiation Area Monitor (RAM) and Continuous Air Monitor (CAM) Calibration at TRA, TRA-008, TRA-1151, Engineering Design File, 05/96 Rev. #02.

"The old Tracerlab RAMs have several calibration and readability problems, and operability is marginal due to lack of parts. National Institute of Standards and Technology (NITS) traceable information with uncertainty evaluations on the RAM calibration fields is not available or is lost due to poor record keeping. Local efforts to provide such values did not include uncertainty evaluations. The old CAMs were calibrated with the incorrect sources using gamma activity rather than beta activity." [Pg.1]

"Because of delays in implementing the new system, the older system is still in use and still connected to the ATR Plant Protection system (PPS) and thus referenced in Technical Specifications. Because of the problems inherent in the old systems, the ATR should expeditiously complete the transition to the new Digital Radiation Monitoring Systems (DRMS)." [pg. 2]

[*It must be noted that these documents show a long standing known problems(inaccuracy of radiation monitoring equipment/data reported) that the more current documents show have yet to be corrected.]*

7. Occurrence Reports, After 2003 redesign ending in 2004. Response to Request No. 8a.

[These were un-reviewed by EDI at this time]

8. Occurrence Reports 6/10/04, Response to Request No.6a.

[These were un-reviewed by EDI at this time]

9. Southeast Safety Rod Failure, INL, Interoffice Memorandum, 2/17/05, from D.G. Robinson Response to Request No. 4i.

"This attempt to manually withdraw and insert the [Safety Rod] SR proved that the problem was in-tank. The problem was likely debris of some kind caught in between the safety rod and the inner or outer snubber tube and/or possibly debris on the safety rod rack tube. Problems of this nature have been experienced in the past with the safety rods."

[*Safety rods also called control rods are crucial to safe shutdown of ATR reactor in an accident (scram) and therefore pose an ongoing safety issue. There is no indication that this problem has been adequately corrected. Also, this is a systematic problem with ATR's "serpentine" fuel/control/safety rod configuration unlike conventional reactors that use straight configuration of fuel/control rods.]*

Attached to this Response to Request No. 4i. Unanswered Safety Question USQ Process Proposed Change Form dated 6/10/04] "The TRA Compressed air system dryers and filters are over thirty (30) years old and are nearing their end of life."

10. Supplement A, Summary Report – Engineering Evaluation, Inspections of the Idaho National Laboratory, Advanced Test Reactor, TRA-670-M-3, TRA-670-M-5, and TRA-670-M-85 Primary Coolant System Heat Exchanger Tubes, Heat Exchanger Shell and Internals, Performed Spring 2007, Prepared by John Fox (9/18/07), Reviewed by James Rodgers (9/18/07). Response to Request No. 9e.

This report was buried in the back of a 2 inch stack of older reports that were not initially reviewed because they were dated.

"The reason for the tube plugging in M-3 was to address 7 tubes that were un-inspectable because of occlusion [obstructed] of the tubing that was sufficiently severe to prohibit the passage of the eddy current probe." [pg. 2]

“Table 5 – Ultrasonic testing results summary primary coolant heat exchanger shells (% of normal) shows UT Maximum Measured for M-3 at 111%, M-5 at 107%, and M-85 at 105%. “ [page 11]

“Considerable amounts of encrusted corrosion product and bio-fouling were noted on the carbon steel surfaces (shell, support plated, tie rods and structural shapes) of the heat exchanger. The amount of this material impeded access to the heat exchanger and limited the video probe and fiber scope inspection.” [pg. 19]

“During the eddy current portion of the examination an incidental finding of cracking in the divider plate on the M-85 heat exchanger and a defect in the cladding were noted… the visual appearance of the divider plate crack suggests that the cracking has been driven by fatigue, as opposed to stress corrosion cracking process. The visual appearance of the clad defect suggested a mechanical or ductile tearing mode and does not suggest an active corrosion cracking process.” [page 20]

“The results of the shell inspection indicate that active corrosion is occurring in the shells. It is likely that the shell will continue to have periodic leakage from pitting and require repair. The most likely areas for through-wall penetration is along a vertical band 180 degrees (opposite) from the centerline of the secondary coolant nozzles. Corrosion to the pint of penetration of the support plates will likely, if not already, occur – just as penetration of the vessel shell has occurred.” [pg. 23]

Life extension of the vessel and the vessel internal structures is jeopardized by biological fouling and biologically influenced corrosion.” [pg.24]

11. Information on the Advanced Test Reactor (ATR) Primary Coolant System (PCS) to Secondary Coolant System (SCS) heat exchangers, Interoffice Memorandum, 4/21/05, From Phillip A. Erickson, White Paper . Attached to Response to Request No. 9e.

This report was buried in the back of a 2 inch stack of older reports that were not initially reviewed because they were dated.

“The design life of heat exchanger is 20-years, reference. The original carbon steel corrosion allowance was 0.125 inches, which equates to 0.00625 inches/year over its design life of 20-years.” [pg. 1]

“[T]hese heat exchangers are at 200 % of their original design life. Replacement of the 670-M-2/3/4/5/85 heat exchangers should be initiated before exceeding the 200% of their design life in 2005.” [pg. 7]

12. Greater-Than-Class C Low-level Radioactive Waste and DOE Greater-Than-Class C-Like Waste Inventory Estimates, USDOE, July 2007. Response to Request No. 9c.

“Total stored GTCC LLW and DOE GTCC-like Waste Volume and Activity Estimates (Table ES.1); Total INL stored and projected volumes are (5,500 cubic meters) with radioactive activity in millions of curies (MCi) 140 [140,000,000 curies]. Activities for the projected inventory estimated at six years following reactor shutdown. Activities for the stored inventory have been decayed to 2007." [Pg. v]

"In addition to estimates of the total volumes and activities for the stored and projected (2035) inventories, the estimated volumes for GTCC and GTCC-like mixed waste which may require treatment prior to disposal ." [pg. 7-6]

"DOE GTCC-like waste, which includes items similar to GTCC LLW under DOE regulation, which may not have a path to disposal." [pg. 7-1]

The disposition of this waste is under review in current DOE GTCC EIS. NRDC lost a legal challenge to maintain the previous designation of this high-level waste and the now arbitrary DOE's new designation of GTCC and GTCC-like waste as low-level waste. Much of this INL waste comes from high-level waste tank remediation operations. This re-classification of formerly high-level waste has major implications on INL high-level waste tank remediation and the ongoing incineration of high-level tank waste operations at the INTEC High-level Waste Evaporator. Robert Bullock who heads IDEQ Hazardous Waste Division sent Broscious an email 3/4/06 that states: "The [INL] reactors are not currently regulated as Treatment Storage or Disposal Facilities." Therefore there is no state oversight on these waste issues.

13. ATR Seismic Safety Basis Deterioration, Potential Inadequacy in the Safety Analysis (PISA) Form TRA-USQ-2004-214 + Rev.1, 6/10/04. Response to Request No. 6a.

"The status of seismic Probabilistic Risk Assessment (PRA) studies for ATR does not adequately support the safety basis." [pg. 1]

"The seismic PRA results from 1991 (and 1994) do not accurately reflect the facility seismic safety posture, and without restoration of commercial power prior to depleting above ground emergency makeup inventories, the probability of seismically-induced fuel damage is increased." [page 2]

14. ATR Seismic Primary Coolant System Break Size and TRA Support Vulnerabilities, Potential Inadequacy in the Safety Analysis (PISA) Form, TRA-USQ-2004-396, 6/10/2004.

"[T]he Advanced Test Reactor Seismic Probabilistic Risk Assessment (PRA) Final identified several smaller Primary Coolant System (PCS) lines that would be vulnerable to breakage during PC-3 or PC-4 seismic events. The seismic report also identified the bypass demineralizer shielding block partition wall as not satisfying ATR seismic criteria for a PC-4 seismic event. Also TRA masonry block buildings constructed in the early 1950's have not been shown to be adequately reinforced and block wall collapse would be expected for PC-3 and PC-4 seismic events. The November 29, 2004 letter report from ARES Corporation also identified inadequate reinforcement of some [Test Reactor Area] TRA [part of ATR] masonry block buildings which has resulted in very low seismic capacities. The likely-hood of masonry block wall collapse, including the TRA-619 building that contains two safety related emergency firewater injection pumps, has not been estimated. Wall collapse in buildings that do not contain safety related equipment could also degrade the emergency firewater injection system inventory and flow capacity due to flow lost from damage to fire water risers in the buildings." [pg. 1]

"Potential for an increase in probability or consequence of an accident or malfunction evaluated in the safety basis. Could the PISA increase probability of occurrence of an accident previously evaluated in the safety basis? YES." [pg. 3]

Mark Sullivan cited this issue in our Complaint. It is restated here to document DOE/ID's acknowledgement in its PISA and more importantly that this fundamental structural problem has no quick fixes nor has any been implemented.

15. ATR Seismic Primary Coolant System Beak Size Contribution from Letdown Valves, Potential Inadequacy in the Safety Analysis (PISA) Form, TRA-USQ-2004-413, 6/10/2004.

"Seismic loss-of-coolant accident events are represented as design basis events in Chapter 15 of the ATR Upgraded Final Safety Analysis Report (UFSAR). No letdown flows from PCV-1-1 or LCV-1-3C were included in the analyses for evaluating thermal margins for the postulated breaks. The omission could increase the potential seismically-induced leakage when combined with seismically-induced pipe breaks, and could result in larger than previously estimated seismically-induced PCS leakage and a reduction in previously calculated thermal margins. Could the PISA increase probability of occurrence of an accident previously evaluated in the safety basis? YES." [pg. 1 & 2]

16. Impact of Potential PCS Leakage on Fuel Element Thermal-hydraulic Conditions Prior to Reactor Vessel Venting, Potential Inadequacy in the Safety Analysis (PISA) Form, TRC-USQ-2005-173, 6/10/2004.

"The safety basis does not include analysis of a complete loss of flow accident with the PCS leakage. The event analyzed assumes the PCS has no leakage and thus pressure remains high throughout the event."

"A scoping calculation, see attached figures, shows severe fuel clad temperature excursions [increases] could occur prior to reaching the elapsed time for vessel vent valve vent actuation if the PCS pressure decreases." [pg.1]

"Could the PISA increase probability of occurrence of an accident previously evaluated in the safety basis ? YES." [pg.2]

17. M-1 Emergency Coolant Pump Flow Measurement and Uncertainty, RTC-USQ-2005-197, 6/10/2004. Potential Inadequacy in the Safety Analysis (PISA) Form.

"Sufficient emergency coolant pump (ECP) flow is required to meet the [ATR reactor] thermal margins established in the ATR Documented Safety Analysis (DSA). The new information is concerned with the DC battery backed ECP M-11. Modeling of the reactor response to a small break loss of coolant accident (SBLOCA) with a loss of commercial power is…[only] during the first 30 minutes. In accidents modeled, the float change would be unavailable from the charger and battery voltage would be reduced with a corresponding reduction in available flow. The above two discrepancies result in the following concern. First, the SBLOAC with a loss on commercial power would also result in the loss of the float charge to the utility battery bank. Overall, a reduction in voltage across the utility battery bank results in a reduced flow from the M-11 ECP." [pg. 1]

"Applying the above limits to this potential in-adequacy reduces the amount of energy that t must be removed from the reactor core following a SBLOCA with a loss of commercial power." [pg.3]

18. Reactor Scram – High Inlet Pressure, Occurrence Report (After 2003 Redesign), NE-ID-BBWI-ATR-2004-007, 8/5/2008. Response to Request No. 7 b.

"On July 10, 2004, the Advanced Test Reactor (ATR) scrammed due to failure of the running diesel generator, due to low lube oil pressure. The backup diesel generator started automatically and assumed the loads. During the process of restarting the ATR 36 minutes later, the reactor again scrammed due to a high primary coolant system inlet pressure. The second scram placed the reactor in a xenon-precluded startup condition. On July 11, 2004, after the time necessary for the xenon to decay, the reactor was returned to full power operation." [pg.6]

19. Crack Found in the ATR 40-Ton Crane Trolley Support Beam, Occurrence Report (After 2003 Redesign), 9/22/04, NE-ID-BBWI-ATR-2004-0010, 8/5/2008. Response to Request No. 7 b.

"During the inspection, QA personnel identified one new weld crack in the trolley support webbing, in a position that mirrors the first crack found the week of June 21, 2004." [pg.2]

20. Advanced Test Reactor (ATR) Reflector Safety Analysis, T.A.Tomberlin, Internal Technical Report, PG-T-89-018, July 1989, Response to Request No. 4 n.

"The ATR reflector, composed primarily of beryllium, experiences significant material changes as a result of its prolonged exposure to the intense neutron and gamma ray environment during reactor operation. Diffusion of the gaseous atoms to dislocation centers can introduce stresses which lead to deformation and failure. Historically, the aging of beryllium in the ATR has resulted in bowing, cracks, small holes through the thin ligaments, and spalling. Aging effects beyond those analyzed to be acceptable constitute reflector failures." [page 1]

"The initial evidence of reflector aging is expected to be progressive cracking leading to a full length crack of the A ligament. However, the potential for other aging effects exists. [P]iece of beryllium against fuel plate 19 (complete [coolant] flow blockage) and complete cracking of the A and B ligaments resulting in a free 'nosepiece' of beryllium." [pg.11]

"Consequences of Inadequate Fuel Element Cooling; Complete blockage of channel 20 by a piece of beryllium is highly improbable but has been evaluated. The consequences of a partial blockage of channel 20 by a piece of beryllium, which is feasible, are addressed in Section 6.12.0 of this report. **Complete blockage of channel 20 can obviously compromise the cooling of [fuel] plate 19 and cause the temperature of the plate to increase significantly**. The most severe conditions would naturally result from blockage in the high-power fuel element positions…" [emphasis added] [pg. 22]

"Three major types of beryllium pieces resulting from ligament cracks were described in section 5.5.1 above. Contact of a [beryllium] nosepiece with an Outer Shim Control Cylinder (OSCC) could block approximately 115 degrees of the OSCC coolant annulus. **Blocking 115 degrees of the outer cooling annulus would represent a blockage of approximately 21%.** " [emphasis added] [pg. 35]

"Partial melting of [fuel] plate 19, if any were to occur, would be limited to the 13-in. length. A 13-in. length of [fuel] plate 19 is about 1.8% of an element and represents considerably less than 1% of the total core."

"Consequences of Fission Product Release to Primary Coolant System; [A] release to the primary coolant system (PCS) of one percent of the core fission products has been considered. An ORIGIN2-calculated fission product inventory for the ATR core after 60 days of operation at 25 MW indicates that a release of 1% of the inventory would be approximately 2.4×10^{6} curies of solids (Cs, Rb, Ba, Te), 1.0×10^{6} curies of halogens (I, Br), and 1.0×10^{6} curies of noble gases (Xe, Kr). [total 440,000 curies] All released fission products would be expected to go into ionic solution with the water. Assuming complete mixing with the 73,800 gal. of primary coolant, the resulting activity level would be 1.6×10^{4} micro-curies per milliliter. This concentration would exceed the technical specifications limit of 20 micro-curies per milliliter and would therefore require shutdown of the reactor. **Release of 1% of the core fission products into the PCS could result in significant releases from the ATR stack**. Efforts would be made, upon experiencing a fission break, to control the immediate stack release rate to less than 400 Ci/day." [emphasis added] [pg.25]

"A stack release of 10^{5} [10,000] Ci/day has been calculated to result in a maximum daily dose of 120 mR at a distance of 100 meters downwind from the ATR. The calculated dose would only be 36 mR…" [pg. 26]

"The worst case consequences of beryllium cracking, relative to Outer Shim Control Cylinder (OSCC) rotation, would be total binding of an OSCC pair. Simultaneous binding of more than one OSCC pair is not considered credible. **Even if only half of the OSCCs were operable, the shutdown margin of the reactor with three safety rods, all neck shims, and half of the OSCCs inserted has been calculated to be approximately 4.0$.**" [emphasis added] [pg. 34]

"Cracks in the reflector could lead to pieces of beryllium being washed out of the reflector and into the primary coolant system (PCS). The possibility of damage to reactor or PCS components by these free pieces of beryllium has been assessed. **Components for which the assessment was made include the heat exchangers, primary coolant pumps, primary coolant pump check valves, safety rods, neck shim rods, outer shim control cylinders, and fuel elements."** [emphasis added] [pg. 43]

"There are two places where relatively large beryllium particles could enter a safety rod drive assembly and settle in locations where a potential problem could develop." [pg.46]

"Movement of beryllium chips through a reactor vessel has, at times resulted in chips becoming lodged in the fuel element channels or the channels between outer fuel plates and the reflector. Potential damage to [fuel] plates at the ATR from this type of [coolant] flow blockage is reduced due to the presence of side vents on the fuel elements." [pg.54]

“Earthquakes; The ATR seismic system is designed to shut the reactor down **before** physical damage from a seismic event can cause damage sufficient to preclude safety rod insertion. A seismic event could conceivable cause failure of a highly stressed [beryllium] reflector.” [pg. 76] *[This assumption of shutdown prior to damage is unsubstantiated especially if there are cascading system failures. Although dated, this report shows a long(20-year) history of ATR component aging problems, compromised reactor fuel cooling, fuel melting and significant consequential radiation releases that have yet to be corrected because of fundamental reactor design. Insertion of the safety/control rods is the primary way the reactor is safely shutdown or scrammed. Also see safety/control rod driver and fuel element failure problems in other FOIA documents.]*

THE END