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Preliminary
Comments on the Content of
Freedom of Information Documents
Released by the U.S. District Court of Wyoming

And Other
Documents Related to
U.S. Department of Energy
Advanced Test Reactor Operations
at the
Idaho National Laboratory

Revision XXIII (4/4/10)

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I. Introduction

This Environmental Defense Institute (EDI) report is a preliminary review of U.S. Department of Energy (DOE) reports related to the department's nuclear Advanced Test Reactor (ATR) located at DOE's Idaho National Laboratory (INL) in southeastern Idaho. INL is a sprawling 572,000 acre nuclear reservation created in 1948 by the then Atomic Energy Commission. This report is a part of EDI's multi-year project to evaluate the environmental, health and safety issues related to the ATR operations. [1]

Sections II, III and IV are reviews of DOE Freedom of Information documents (FOIA) released in *Keep Yellowstone Nuclear Free, Environmental Defense Institute and David McCoy v. U.S. Department of Energy*, in U.S. District Court for the District of Wyoming (06-CV-205-D), October 2009. Document references below refer to internet postings of about 1400 pages of official Wyoming District Court documents. [2]

Sections V and VI review DOE documents gained through previous EDI FOIAs and reports posted on DOE's website and/or posted in weekly DOE/INL Operation Summaries.

This EDI review is blocked from including more than 211 pages of DOE censored redactions in this FOIA disclosure that include;

a.) Updated Final Safety Analysis Report (UFUFSAR) – 104 pages;
b.) Emergency Management Hazardous Assessment (HAD) – 98 pages;
c.) Engineering Design File (EDF) No. 4394 – 9 pages. [3]

DOE's censored "redactions" cited by EDI below only relate to those FOIA document sections commented on, and not the total redactions in the whole FOIA release enumerated above. Additionally, DOE is currently using a redaction technique different from the old "blacked out" that is readily identifiable. Now, DOE uses a total "blank-out" that makes identifying redactions extremely difficult unless they occur mid-page where it is discernable.

When redactions are cited, EDI asks the question WHY ? This question is repeatedly posed because there is no apparent/credible "national security" issue; but rather a "national security embarrassment" issue where the DOE intends to block public access to information crucial to downwinders who are subject to radioactive emissions from the ATR operations.

In our litigation, DOE stood fast on FOIA redactions related to operational ATR deficiencies because they could identify weaknesses that could be used "by an adversary to cause release of significant quantities of radioactive material..." Yet they released postulated data on accidents!

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[1] For a more comprehensive analysis see EDI's "Unacceptable Risk, at the Idaho National Laboratory Advanced Test Reactor, The Case for Closure;" available on EDI's website publications.
[2] See; http://ecf.wyd.uscourts.gov/
[3] Plaintiff attorney Mark Sullivan, "Redaction Log." Also see; Broscious Comments on FOIA Plus Specific Redactions (C) 11/5/09.

The difference is crucial. DOE clearly understands that any credible legal challenge to the ATR's "safe operations" requires information about existing/identifiable fundamental facility deficiencies!

Although important in quantifying risk, the postulated accident data are JUST scenarios; whereas, information that can identify crucial design/structural/seismic/operational deficiencies will raise fundamental questions on why this 40-year old ATR is allowed to operate for several more decades given the huge risks to the deliberately uninformed public. Despite EDI and KYNF's legal attempt to force DOE conduct a comprehensive Environmental Impact Statement (EIS) on the ATR, the Idaho Federal Court ruled against it. [4] An EIS would have offered a official assessment of the full range of environmental, health and safety issues that the public could comment on.

The Advanced Test Reactor (ATR) located at the Idaho National Laboratory (INL) was designed in the 1950s - constructed in the late 1960s using the regulations applicable at the time - and power operations commenced in 1969. The ATR design is the most complex in the world due to the "serpentine" core fuel design [5] and "clover-leaf" core configuration. Most importantly, the Department of Energy's (DOE) primary mission for the ATR is testing of new reactor fuel for the U.S. Navy's Nuclear Propulsion Program, military/NASA production and commercial power reactor applications. This testing of fuel - fuel cladding (material that encapsulates the uranium) types is intended to simulate "real time" neutron exposure to evaluate how the fuel/cladding withstands actual reactor operating conditions. Consequently, there is a long history of experimental fuel failures that by definition result in excess emissions and significant vulnerabilities to accidents. Add to this, completely separate fundamental ATR system inadequacies due to design complexity and aging; and you see a major accident waiting to happen.

Admittedly, the external demands on DOE to continue ATR operations are significant. The Navy does not allow untested reactor fuel used in its nuclear submarines and/or aircraft carriers. NASA also demands tested nuclear fuel as well as production of plutonium-238 nuclear fuel for its space power program requirements. There are other minor missions – such as medical isotopes production.

"The ATR has a cloverleaf arrangement of aluminum clad, plate type fuel elements which result in flux traps for experiment irradiations. The ATR also has a combination of rotating shim drums outside the fueled area with small diameter shim rods inside the fueled area which allow operation with [nine] different power (flux) levels in different segments of the core." [6] This means that power flux levels in one of the nine flux tubes can be at 200 **K**W (200,000 Watts) and simultaneously 431 **M**W (431,000,000 Watts) in another flux tube. This range in concurrent power levels in different lobes of the reactor means equally different heat levels and coolant level requirements.

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4 KYNF-EDI at al. v DOE; Idaho Federal District Court, Civ. No. 07-36-E-BLW.
5 Commercial nuclear power reactors have a straight forward vertical fuel core and control rod configuration.
6 UFSAR-153, pg. 15.4-4

This degree of extreme design complexity – major variation in concurrent reactor core power levels - the ATR would never be allowed as a permitted commercial power reactor due to the difficulty controlling reactor power levels/different coolant within the core and concurrent potential vulnerabilities inherent in a self-regulated 40 year-old reactor that is long past its 20-year design life.

DOE's own previous Environmental Impact Statement on NASA's plutonium-238 production states: **the ATR released 1,802 curies in 2000 and 1,180 curies in 2003 to the atmosphere.** [7] On average that is about 1,491 curies/year; so over an eight year period 2000 through 2009 (given ATR's continuous operation) about 13,419 curies may have been released to the air. These high emissions from ATR suggest liquid waste is first sent to the ATR cooling towers w/o treatment and the precipitates are then pumped to INTEC evaporators or the percolation ponds. This represents a significant hazard to INL workers and the downwind public, and violations to regulatory limits discussed below in Section III.

Summary

These EDI comments – as the title shows are preliminary, and are not comprehensive but are a cursory review of an ongoing effort to analyze the content of the subject DOE documents. That said, and despite extensive DOE redactions, there exists in these comments sufficient revelations for the public and regulators to be alerted to the significant public hazard continued ATR operation poses.

DOE intends continuing ATR operations through at least the year 2040 in its "Life Extension Plan (LEP)." Plaintiffs filed a separate lawsuit in 2007 with the U.S. District Court of Idaho [8] requesting that an Environmental Impact Statement be conducted by the DOE on this ATR-LEP program. The Court ruled in favor of DOE in this case and thus blocked the publics' right – normally provided under the National Environmental Policy Act – to a comprehensive assessment of the environmental consequences of past/future impacts of ATR operations.

This recent Wyoming District Court Order forcing DOE to release crucial documentation on ATR operations represents a significant step towards the publics' understanding about the ATR. However, it does not carry the same "official" comprehensive assessment or the opportunity for "official" public comment of an Environmental Impact Statement. The bottom line is DOE continues to use every political/legal devise available to prevent the general public from knowing the truth about the hazards of continued ATR operations.

Moreover, Plaintiffs legal efforts took five years and significant resources just to force DOE to release these 2004 documents that are now five years old and outdated. This is unconscionable by any standards of "openness in government" touted by the current Obama Administration. Plaintiffs are now faced with the new legal challenge of a new FOIA for the current ATR safety documents that could take another five years !

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[7] DOE/EIS-0287 pg. 4-30; DOE/DEIS-0373D, pg 3-26.
[8] KYNF-EDI at al. v DOE; Idaho Federal District Court, Civ. No. 07-36-E-BLW.

Section II. 2004 Chapter 15 Severe Accident Analysis – Upgraded Final Safety Analysis Report (UFSAR-153) for the Advanced Test Reactor (8/10/04)

Identified below by document ascending page number (15.*-*);
Document text excerpts are in "quotation marks" and **Bold** only emphasizes original text font.

15.0-12 and 13; "Condition 4: Extremely Unlikely Faults" has two paragraphs **Redacted !**

15.1-2; "Excessive Heat Removal From the Secondary Coolant System; The secondary coolant heat load is dissipated to the atmosphere via a coolant tower."

This above fact is important because of documented leaks in the heat exchangers and other discharges that as previously cited in above Section I, the ATR released 1,802 curies in 2000 and 1,180 curies in 2003 to the atmosphere.

15.2-1; "The SCS [secondary coolant system] is the principal heat removal system for the ATR. Increasing the secondary temperature or decreasing the secondary flow rate will result in a decreased heat removal rate for the ATR. A break in the secondary piping with a resultant loss of secondary coolant inventory from the primary heat exchangers is a severe example. The secondary temperature and flow can also be influenced by failure of support systems such as instrument air. Oxide or crud buildup in the heat exchanger tubes can also decrease the rate of heat removal."

15.2-2; "Catastrophic failure of the piping between the secondary pumps and the heat exchangers would result in complete loss of flow to the heat exchangers with a rapid depletion of the coolant inventory from the secondary of the heat exchangers. The combination events would result in essentially no heat removal from the primary coolant….As the temperature of the primary coolant continues to rise due to the decay heat; the volume will increase, resulting in a pressure rise. The pressure relief valves will maintain the pressure below damage levels."

Any failure of the SCS will have an immediate impact on the primary coolant system (PCS)'s ability to maintain safe operating temperatures and therefore be a potential cascading event triggering other system failures. See 15.13-2 below on vulnerability of PSC check valves.

Additionally, "The ATR PCS/SCS heat exchangers are operating beyond 200% of their 20-year design life." "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 & corrected so it can be expected to fail two to four years from now."** [Emphasis added][pg. 3] [9]

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

15.3-4; "Long-Term Complete Loss of Flow; **The probability that both diesel and commercial power will fail simultaneously is relatively high. However, the probability of restoring one or both sources within 30 minutes is also relatively high** (Thatcher 1997) If the battery-backed power is a complete loss of [coolant] flow in less than 30 minutes is beyond design basis; the complete loss of flow after 30 minutes is estimated to be a Condition 4 event (Thatcher 1997)." [Emphasis added]

These above critical safety system vulnerability disclosure statements are buried here, apparently never to reappear in DOE's conclusions as the "likely" extreme hazard they represent.

15.3-5; "However, heat losses through the vessel walls are not sufficient to dissipate the total decay heat, and the coolant in the reactor vessel will heat up. At this point, the transient is similar to the complete loss of secondary (Section 15.2-1) in that pressure will rise, the relief valves will cycle and depressurization will be required if forced circulation is not restored before the vessel coolant temperature reached approximately 200 degrees F."

15.3-16 and 17; Figures 15.3-5 and 6 graphically shows near vertical and off-chart spike of CHF [critical heat flux] during loss-of-commercial power accident, two pump operation after only 1.3 seconds.

15.4-1; "The ATR has three reactivity control systems which are used to control and shut down the reactor. These systems are: 1.) Outer shim control cylinders; 2.) Neck shim rods; 3.) Safety rods (activated by the PPS [primary pump system]….The safety rods [also called control rods] are the only reactivity control elements modeled in the analysis to terminate power transients [reactor power spikes]….Full withdrawal of the safety rods requires about **20 minutes** when the timer is controlling the withdrawal….Perturbations of the neutronic [sic] balance in the reactor core will result in an increase or decrease in reactor power….Larger perturbations will result in a **reactivity initiated accident since the regulation rod cannot compensate for the insertion.**" [Emphasis added]

The above discloses the uniquely complex reactor power control systems (each of which has their own vulnerabilities – discussed below. Additionally, the **20 minute time** required for the safety rod insertion radically contrasts to the 3-5 seconds for power excursions discussed below. This is in contrast to commercial power reactors have relatively simple power control systems.

Note below; In a ATR fuel melt-down; "The initial temperature of the relocating material [molten ATR fuel debris] was assumed to be 1250 K [Kelvin] [976.84 Celsius], a conservative estimate for [fuel] melt held up in the core a few seconds after melting within **3-5 seconds of scram**." [15.12-17] This difference between **20 minutes** for safety rod insertion and the **3-5 seconds** for fuel melt represents a crucial hazard/deficiency in the ATR safety systems ability to respond to reactor power excursions/transients/power spikes.

15.4-9; "Conclusion – The protective criteria margins will be preserved with maximum effective Plate 15 powers of 443 MW and 417 MW got for the three-pump operation and the two- pump operation, respectively. However, as is evident from the time scale involved, the dominate effect in terminating the event is inherent feedback. "

In other words, if the "feedback" instrument systems are not accurate; therefore, there are significant problems – which is in fact a cited vulnerability.

15.4-12; "Results – The RELAPS5 results show a peak reactor power of **435 MW occurs at 0.04 seconds** for the two pumps and 369 MW at 0.04 seconds for three pumps."

Crucial to effective ATR scram is the insertion of reactor safety control rods that have a history of degradation and failure. [10] Despite the hazard, DOE views "The unique capability of the ATR to provide either constant or variable neutron flux during a reactor operating cycle makes irradiations in this reactor very desirable." [11]

During startup of the Advanced Test Reactor on March 8, 2009, it was determined that a primary coolant check valve was not seating properly. Startup preparations were stopped, the primary coolant system was depressurized and the reactor was defueled so the check valve could be replaced. (NE-ID-BEA-ATR-2009-0003).

15.4-12 – Section 15.4.5.2.3, "Conclusion – The protective criteria margins…" has half page + 2/3 of the following page **redacted**. WHY?

15.4.7.1; "Rapid Regulating Rod Withdrawal. A component failure can cause the regulating rod to drive out. This event can occur at power levels between N_L and N_F " or [321.6 MW reported in Section 15.4.2.2 and far below the 443 MW power level reported in Sections 15.4-9 and 15.15]. This crucial ATR safety system is more at risk of failure due to current reactor power levels than DOE is willing to disclose; OR, if disclosed it was redacted as noted above.

As cited below, the ATR power level of 431 MW is nearly twice the DOE ATR power level claim of 250 MW. [See Facility Certification Report No. 29 by Battelle Energy Alliance catalogues equipment failures and malfunctions due to age of the ATR and unavailability of replacement parts.]

15.4.7.2; "Withdrawal of all outer shims…failure of a timer interlock can occur due to a single failure in the Log N system." "Withdrawal of all outer shims and One Neck shim from N_L …this fault can occur as a result of operator error. The maximum addition rate for withdrawal of a single neck shim id 0.2c/sec."

15.4.7.6; "Loop Flow Coast-down or Loop Loss of Temperature Control with a Loop

[10] Occurrence Report, NE-ID-BEA-ATR-2006-0009, #3 Safety Rod Actuator Controller Failures. Also see Occurrence Report, NE-ID-BEA-ATR-2007-0001.

[11] Frances M. Marshall Advanced Test Reactor Capabilities and Future Operating Plans, September 2005, INL/CON-05-00549, Idaho National Laboratory.

Instrumentation Initiated Reactor Trip….this failure is initiated by a low of commercial power (or other events affecting the loop pumps) of failure of the temperature control due to controller failure or operator error."

15.4.7.7.1; "Power Axial Locating Mechanism Drive System Failures; Simultaneous movement of two PALM tests can occur from several sources, e.g., a high-temperature up-drive that occurs in one PALM while the other is cycling normally, or a control system failure."

15.4.7.7.2; "Uncontrolled Movement of a Positioner [sic] Test; However, failures of the hardware or mechanism or operator failures could result in large movement with reactivity insertions that are outside of the control range of the regulating rod. Additionally, if the test is withdrawn to the upper limit, the configuration could represent and unacceptable potential void reactivity insertion."

15.4.7.8; "Withdrawal of All Outer Shims and Neck Shims; Failure of control rod withdrawal timer interlock along with operator error, can result in withdrawal of all outer shims and neck shims…This results in a reactivity addition rate of **7.9c/sec.."**

15.4.7.9; "In-pile Tube Voiding Due to Opening of a Normally Inaccessible Valve; is listed as one of the mechanism Drive System failures."

15.4.7.11; "Loss of Loop Temperature Control Due to Heat Exchanger Failure or Line Heaters Sticking on Without Loop Scram; This fault is initiated by temperature control valve failure, controller failure for heat exchanger flow or line heaters, or loss of secondary [coolant] flow. This causes a gradual temperature rise (3 degree per second maximum) in the loop coolant with a resultant decrease in [coolant] density…. Increase the power slowly to the neutron level PPS [Primary Pump System] subsystem set-point and cause a scram." Other DOE reports state that these heat exchangers are 200% past their design life and that they leak.

15.4.7.12; "Slow Lobe Power Balance Shift Due to Shim, Lobe Power Indicating System or Operator Failure with Operator Compensation. The regulating rod is generally maintained within the operating range by rotation of shim drum pairs. Erroneous movement of shims in or adjacent to high power love could occur as a result of shim failures or a failure of the lobe power shims, above powers exceeding the PPS neutron level subsystem or WRS scram set-points mat be reached. This assumes the PPS neutron level subsystem and WRS channels adjacent to the high power lobe are functional."

15.4.7.16; "Rapid Regulating Rod Withdrawal with Failure of the Wide Range Subsystem. The failure of the regulating rod is discussed in Section 15.4.7.1. The envelope discussed assumes that the event is terminated by the feedback and the high power scram. There is no consideration of a scram on rate of change (WRS function); therefore, the evaluation for the high probability event (Condition 2) is applicable to the event with the failure of the WRS."

15.4.7.17; "Withdrawal of all Outer Shims and Neck Shims Coupled with Rapid Regulating-Rod-Withdrawal; This fault is a combination of the faults described in Sections 15.4.7.1 and 15.4.2 with concurrent failure of the shim withdrawal permit interlock when the reg-rod is below 20 inches withdrawal. The combined accident is a ramp of .079$/sec … then an unlimited ramp at **1.079/sec**."

15.4.7.18; "Driven Test Loop Blow-down Experiment Hardware Failure."

15.4.7.19; "Voiding in All In-pile Tubes Due to a Simultaneous Flow Coast-down or Loss of Temperature Control in all Loops. Failure of the commercial and diesel power sources will result in simultaneous coast down of all loop pumps. If the loop scrams fail, voiding in all IPTs could occur. Loss of heat sink (i.e., failure of the HDW system) to the experiment loops or loop heater failing on without a loop scram could also cause voi8ding in all the IPTs. The loss of temperature control (event titled "Voiding in all IPTs due to a simultaneous loss of temperature control in all loops") progresses slower than the coast down and is enveloped by the coast down analysis. Several passes are required for heating in the loss of temperature control; the flow coast down occurs in less than **3.5 seconds**. Concurrent failure of the loop protective system is assumed, so no reactor scram will occur until the neutron level PPS subsystem trip at $1.2N_F$." N_F = 250 MW power; 1.2 N F = 300 MW. **Page 15.4-24 graphically shows the extremely rapid rise in transient power level from 305 MW to 322 MW in 3 seconds.**

15.4-29; Figure 15.4-7 graphically shows the critical heat flux, flow instability in Standard In-pile Tube going off the chart in **35 seconds**.

15.4-37; Figure 15.4-15 graphically shows "Reactor Power (SIPT [standard In-pile tube] pump discharge pipe break RIA [radioactivity insertion accident] –two-pump operation" power spiking to **525 MW in 10 seconds**.

15.5-4; "A loss of Instrument Air Closes PCV-1-1 and Opens FCV-1-8 with concurrent Failure of the Inlet Pressure Subsystem. This fault is initiated as discussed in Section 15.5.1.1 but with concurrent failure of the inlet pressure PPS [primary pump system] subsystem."

15.5-6; Figure 15.5-1 graphically shows Primary Coolant System 460 psi pressure spike going off chart in **10 seconds.**

15.6-1; "Decrease in Primary Coolant Inventory; Excessive bleed or discharge from the PCS pressure boundary will result in a primary coolant inventory decrease. This may occur as a result of mechanical failure…" The rest of this section is **redacted** ! Why

15.6-2; "[Seismic] Event M [very small seismic LOCA (loss-of-coolant accident)] could be considered either a Condition 3 or a Condition 4 event, depending on the size of the seismic event leading to the damage. … [Seismic] Events C-E, I, J, L, M or N during high power or maximum plate powers- Condition 4." Other sections of this part are **redacted !** Why

15.6-3; "Break Spectrum Study; Direct core damage is a consequence of an ATR accident for which significant core fuel damage may result as a direct consequence of the accident sequence without the possibility of successful mitigation of the event by the current ATR Plant Protection System (PPS) or Engineered Safety Features (ESF) or the need for additional system failures to occur before fuel damage may result. Some events can and will lead to direct but local fuel damage, such as blockage of fuel cooling cannels by foreign debris, or medium or large break LOCAs in non-sensitive piping and occurring at moderate power levels. These events are not considered to be a direct core damage event as long as the consequence of such direct local fuel damage events meets the required protection criteria for Condition 4 events."

Although revealing, this is a preposterous statement unsupported by previous UFSAR citations !

15.6-5; "Heat Exchanger Tube Rupture; Reactivity can be introduced into the SCS [secondary coolant system] by leakage from the PCS [primary coolant system] through heat exchanger tube leaks. Radioactivity in the SCS **can be released to the environment via the cooling towers."**

15.6-6; "The calculations assumed continuous operation with radioactivity leaking from the PCS to the SCS and maximum SCS cooling tower blow-down. Radio-isotopic mixes representative of normal operation with a fuel element cladding defect (fission break) were considered. Velen's [sic] analyses concluded that the continuous releases with SCS radioactivity concentrations of 0.01 uCi/ml would result in doses of no more than **2.5 mrem/year EDE [effective dose equivalent] to full time resident at the INEEL NSB [nearest site boundary] and no more than 500 mrem/year EDE [effective dose equivalent] to maximally exposed TRA worker.** These values were based on DOE Orders and guidance in effect at the time of the analyses [1989]."

15.6-9; "Upper Vessel or Bottom Head EFIS Out-of-Service" section has 3 inch and 5 inch **redaction on the following page.** WHY?

15.6-10; "Piping connected to the PCS that is less than 3 in. in diameter ore separated from the PCS by an orifice with a diameter of 3 in. or less (or a closed valve or check valve has not been seismically analyzed. A severe earthquake, however, may cause some cracking of this piping."

15.6-12; "Conclusions- The seismic level subsystem will terminate reactor operation and , in conjunction with the battery-backed power systems, maintain Condition 3 and 4 protective criteria for the 2 and 3 inch seismic LOCA, respectively. However, as noted above, FI margins following the loss of DC power were lower for the 2 in. LOCA than the 3 in. LOCA. This led to a concern that a very small seismically-induced pipe break could potentially result in unacceptable safety margins." "This event could occur with either a Condition 3 or 4 frequencies." "Results – Nominal and worst-case fuel plate centerline temperatures following the loss of DC power are shown in 15.6-18. In each case, substantial boiling occurred in coolant channels 19 and 20, and the critical heat flux was occasionally exceeded."

15.6-14; "Activities with the ATR shut down occur on a regular basis. Even though the thermal power of the irradiated fuel due to radioactive decay will become less than 0.01 N_F [250 MW] after 90 minutes, a potential for fuel damage still exists chould the vessel be drained to below the top of fuel in the core due to a LOCA during outages. A LOCA initiated by a failure in the bottom head can be a difficult sequence to mitigate."

Section 15.6.10.1; "Opening Valve Outside Radiographic Limits or a Radiographic Boundary" has three large **redactions**. WHY?

15.7-1; "Radioactive Release from a Subsystem or Component; The pressurized water loop facilities contain tests with a significant inventory of radioactive material. The radioactive material contained in the tests could be released wither by failure of the loop piping or by dropping of a test out of the cask during handling…**whose failure would result in the uncontrolled release of radioactivity to the environment under transient conditions… A loss of flow or a loss of coolant accident (LOCA) in the experiment loop could result in melting of the test.**" [Emphasis added]

"Worst case (99.5%) site specific meteorology was used. **The resulting dose was 192 mrem, TEDE, for the maximally exposed individual at the INEEL**….Since operating personnel are routinely in areas of the plant close to the loop facility cubicles, the potential personnel exposures resulting from the loop facility LOAC were determined. The dose for a worker outside of the loop cubicle including inhalation and external dose was determined. The analysis assumed the worker evacuated the area in 5 minutes in response to radiation and air monitoring equipment alarms. **The resulting dose was 1.95 rem, TEDE**… The dose to other TRA personnel was 7.4 mrem, TEDE. The dose to workers at INTEC was 15.8 mrem TEDE. INTEC doses are higher due to a longer evacuation time…. The dose for a 200-KW experiment was 192 mrem…". [Emphasis added] This 200 **KW** is a thousand times below the 250 **MW** to 431 MW power levels reported below.

15.7-2; "Radioactive Release from a Subsystem or Component; The worst case setting is 350 mR/**hr** (Peterson 1995). With a response time of three minutes for the action to seal [ATR] confinement, the estimated **thyroid inhalation dose at TRA would be 30 rem.**" "For the event where a cask holding an experiment is dropped, the experiment could be ejected. This ejection could lead to a significant direct radiation exposure." [Emphasis added] At this hourly dose rate, what would be the annual dose is ?

EPA 40 CFR 61.92 standards is 10 mR/yr or 0.01 rem/yr. See Section III below for discussion on units/standards.

15.8-4; "Inadvertent Opening or Failure of a Canal Drain or Drain Piping with the Drain Cover Also Having Been Removed," has a 3 inch **redaction** ! Why ??

15.8-7; "Fuel Storage Canal and Cask Handling Events; Various maintenance activities on the PCS [Primary Coolant System] or systems connected to the PCS pose some threat of draining the PCS…the threat is extended to the canal. Particularly if the bulkheads installed are not full height, the amount of water available can be a concern in the longer term…when the makeup requirements…are modest. ATR fuel elements that are coolable [sic] in air will not melt if the canal drains; **however, direct radiation levels from the uncovered fuel elements within the canal would be severe."** [Emphasis added]

What about the ATR fuel elements that are not coolable [sic] in air ? Will they melt causing an uncontrollable criticality that could boil off the remaining coolant water in the canal ?

15.8-10; "Potential doses due to significant fuel damage and melting in the storage canal (without canal draining) is enveloped by the analysis for the large-break (LOCA) [Loss-of-Coolant Accident]. The dose calculated for 100% core melt considers release of 64% of the source term in the first day and remainder over the next 10 days; **the total dose is 185 rem thyroid and 13.2 rem EDE (whole body)** at the LPZ [low population zone]. The estimated dose at the LPZ is 57.4 rem thyroid and 4.1 rem whole body for failure of eight fuel elements."

This above estimate is based on failure of only eight fuel elements; however, it fails to include the full canal fuel inventory.

"Fuel elements may be out of storage during various evolutions (e.g., while in transient to the reactor, for inspection or for leak testing of the clad) and canal draining events could cause melting of the irradiated fuel elements." Although DOE offers no numbers, this event could release significant radiation into the ATR facility and the environment since the ATR has little or no containment.

15.8.8; This section contains numerous significant ATR vulnerabilities in fuel cask movement in the Canal that DOE considers "Condition 3 Events." This section requires more analysis in addition to structural vulnerabilities of the canal, the canal liner and the canal over-structure.

15.8.10.2 through 15.8.18; "Analysis of Effects and Consequences"; These sections related to cask handling vulnerabilities are heavily **redacted** in more than six sections. Why? Surely, disclosing this information is not a national security risk.

15.9-2; "Momentary Loss of Flow to an Emergency coolant Pump Failure or Inadvertently Turned Off; …is considered a Condition 3 loss of flow event. For this event, operators are assumed to be able to restore flow from at least on ECP [Emergency Coolant Pump] (e.g., restoring commercial and or diesel power **within several hours and before adequate core cooling is lost.** An extended loss of flow event occurring during depressurized operation, wherein operators are unable to re-establish ECP flow, is a Condition 4 event,"

As cited previously, power excursions can occur within 3-5 seconds. Also the power level used in this section (2.5 MW) is not close to the actual ATR power levels of 250-431 MW.

15.9-3; There is a **redaction** of the Trip Initiator emergency coolant flow trip table. WHY?

15.9-4; Extended Loss of Flow Due to Extended Failure of Both Emergency Coolant Pumps; … is considered a Condition 4 loss of flow event." "Identification of Causes and Accident Description; Rapid closing of the butterfly valve can occur by failure of the valve shaft or the valve stop failing. These events are conservatively considered to be a Condition 4 loss of flow event." "Core flow bypass due to a hole in an outlet pipe inside the vessel is discussed as a Condition 4 loss of flow event."

15.9-5; Section "Opening Valve Outside Radiographic Limits or Radiographic Boundary; …that could cause a LOCA during open head operation. … Analysis of Effects and Consequences; has major **redactions**. Why?

15.9-6; "Piping Rupture Outside Radiographic Limits; Identification of Causes and Accident Description – PCS [primary coolant system] piping rupture outside the radiographic limits is conservatively considered a Condition 4 event that could cause a LOCA during depressurized reactor operation."

"The unintentional opening of several drain valves (all of which are outside the radiographic limits), or opening a normally inaccessible or locked valve, is considered a Condition 4 event that could cause a LOCA during depressurized operations."

"Failure of a bottom head closure plate penetration (closure plug), resulting in a LOCA is conservatively considered a Condition 4 event during depressurized operation."

15.11.2; "Analysis of Effects and Consequences – RMSS [radiation monitoring and seal system] damper control circuitry is fail-safe on loss of power, that is, the building exhaust and/or building seal functions will be actuated upon a loss of power to the RMSS system. Upon loss of diesel and commercial power, springs (which are compressed when the dampers are opened) close the dampers. When the dampers are closed, the ATR confinement is sealed, and ther is no threat to the public because…"

The rest of this page and the following pages 15.11-3 and 4 are all **redacted**. WHY ? **DOE clearly will not disclose that this venting system releases radiation to the atmosphere regularly and/or during various accident scenarios.**

15.11-6 through 8; "Momentary Total Loss of HDW [high pressure demoralized coolant water]" have major **Redactions**. WHY ? DOE's use of the word "momentary" sounds innocuous. However, when considered in the context of previous disclosures that show it only takes 1.3 seconds for ATR core temperature to spike off the chart – it has significant meaning.

15.11.11; This page has three large sections **Redacted** before and after "Loss of Diesel Power With Concurrent Failure of the IPT [In-pile Tube] Experiment." Why?

15.11-12; This page has a large section **Redacted** related to "High Pressure Demineralization Water coolant temperature with concurrent failure of the In-pile Experiment coolant instrumentation." Why ?

15.12-1; "Severe Accident Analysis; A high frequency sequence was selected in each of several PDSs [plant damage state] so that the major types of transients and the dominate sequences identified for the ATR would be investigated. The major types of transients are: a CLOFA [complete loss-of-flow accident], LOCA [loss of-coolant accident] -small and large breaks, anticipated transients without scram (ATWS) events, severe reactivity insertion accidents (RIAs), and draining of the storage canal. **While not all of the PDSs have been addressed directly**, those that have provided direction into how the transient consequences and fission product behavior would be characterized for the remaining PDSs." [Emphasis added]

The following paragraph "Core damage occur in the ATR" is **Redacted** ! Why?

15.12-2; "Plant Damage State 1, Low Pressure Boil-off; With no source of water available, the decay heat slowly boiled the liquid in the reactor vessel. The resulting core uncovering, which began at 18.2 h, occurred near atmospheric pressure because the vent valves were open. The fuel melting temperature is reached at 18.9 g. A complete core melt ensured, with a complete release of all the highly volatile fission products (Xe, Kr, I, Cs, Te) in the core."

"The release path for the fission products was from the reactor vessel through vent vales, through the discharge piping to a header connecting to the warm waste tank, then back up through floor drains connected to the header into the second basement. While about 13% of the Te was retained in the reactor vessel by chemiabsorption [sic], nearly all of the remaining fission products were released through the vent valves."

15.12-3; "Plant Damage State 2, High Pressure Boil-off," The first paragraph is **Redacted**.

15.12-4; Plant Damage State 3, LOCA [loss-of-coolant accident] with No Low Pressure

Injection." The first paragraph is **Redacted**. Why? "When the fission products were released a vapor- filled path existed between the fuel and the break. This resulted in virtually no retention of fission products in either the reactor vessel or the piping between the core and the break."

15.12-5; "Plant Damage State 3M, Direct Damage LOCA [loss-of-coolant accident]," first paragraph is **Redacted**. "Fuel melting began near 85 s [seconds] and relocation of molten material to the flow distribution tank fear 174 s, and firewater injection to the vessel began at about 197 s."

15.12-6; "Plant Damage State 4, ATWS [anticipated transient without scram]" first paragraph is **Redacted** ! "Although ATWS events leading to fuel melt have not been analyzed because they are very improbable, the results of the analysis in the preceding sections are used below to indicate the likely behavior of fission product releases resulting from such events."

15.12-8; "Plant Damage State 5M, Direct Damage Large RIA [reactivity insertion accident]; This event is a RIA initiated by experiment loop ruptures that are severe enough to result in fuel melting before the reactor scram and be effective in terminating the reactivity transient. This event category is represented by a single event referred to be the FTVA [flux trap voiding accident]. The FTVA is a bounding, extremely rapid, positive ramp insertion of reactivity in the ATR core due to voiding in a high positive reactivity region of an ATR core flux trap. **The double rupture results in an expulsion of high temperature, high pressure loop water into the relatively low pressure (reactor vessel pressure) flux trap annulus between the gas envelope tube and the flux trap baffle, which will very rapidly void the flux trap annulus. …This rapid positive reactivity insertion and the potential reactivity worth in the flux trap annulus."**

"Destructive reactivity transient tests (SPERT-ID [Miller, et.al. 1964] and the SL-1 accident (AEC 1964) have indicated that a vapor explosion is a possible phenomenon for severe reactivity transients in plate-fueled reactors. [See discussion below on SPERT and SL-1]. The postulated mechanism for the vapor explosion is that the rapid power rise in the fuel plates causes melting and high temperatures in the fuel core of the plates, which results in jets of high temperature molten material being ejected through the weakened cladding into cold coolant channels. **The high temperature material breaks up into small droplets in the coolant, and the resulting large surface area provides for a very rapid generation of steam known as a steam explosion.** The normal pressure limiting mechanisms such as ESF [emergency safety feature] relief valves or other means of transferring water out of the reactor vessel are unable to respond fast enough to accommodate the rapid steam generation and therefore, very high transient pressures **may result in reactor vessel damage."**

"Analyses were performed for a bounding flux trap voiding accident at the ATR. … The analyses calculated that the consequences of this very low probability event are a very rapid positive ramp insertion of reactivity…which results in a peak transient power of about 900 MW in 62 ms [mili-seconds] (Nielson 1990). This extreme transient power is predicted to result in rapid melting of 1.7% of the core…A vapor explosion is postulated to result from the expulsion of the molten fuel into the coolant channels. **The consequences of the postulated vapor explosion is core-wide damage,** but the reactor vessel will not be failed and it will be restrained from excessive vertical movement in response to pressure pulses from the vapor explosion."

DOE's own **"**ATR Reactor Vessel Internals Lifetime Scoping Analysis" report states; "The ATR Aging and Life Extension Program has identified seven critical reactor vessel internal components requiring further evaluation to assess aging. These major components include the core support tank, flow distribution tank, reflector support tank, core reflector tank, inlet flow baffle, thermal shield assembly, and the in-vessel quadrant outlet flow pipe assemblies."

"The seven critical reactor vessel internal components are constructed from various materials. Some of these materials are ASME Code Section III approved and others are **not**. Briefly, the core reflector tank is mainly constructed of the aluminum alloy, while the reflector support tank is a sand casting using the aluminum allow. These two aluminum alloys are **not** ASME Code Section III approved materials. This means that allowable stress values and fatigue curves were not readily available in the ASME Code and had to be estimated." [PG-T-89-011]

Therefore, the above DOE (15.12-8) dubious claim to ATR reactor vessel being "restrained from excessive vertical movement" in the event of an explosion is **not credible**.

Also the above reference to the SPERT tests; included a series of three tests between 1962 and 1964. Other similar tests included SNAPTRAN between 1964 and 1966; and BORAX test in 1954. All of these tests – and there were scores of other tests - of actual reactors deliberately ran them to meltdown/explosion to assess the fuel type and reactor design operating parameters to meltdown/explosion. Millions of curies of radiation was released during these tests and are more fully described in EDI's *Citizens Guide to INL* available on EDI's website.

The above DOE reference to the INL SL-1 reactor accident – occurred in 1961 that killed three operators and seriously radiated scores of first responders and cleanup personnel.

Additionally, the above DOE claim of an ATR explosion is "improbable" is not supported by the preponderance of evidence, long history of ATR fuel failures and DOE's own documented history of reactor tests at INL.

15.12-10; "Plant Damage State 7, Canal Draining, is associated with draining the liquid from the fuel storage canal. There are two major concerns with this event. First, the canal contains a large number of irradiated fuel assemblies, so the potential source term could be very large. Second, the canal is outside of the confinement so that any fission product release from the fuel would have a more direct path to the environment."

"However, for the consequence analysis it is conservatively assumed that all of the released fission products reach the environment within 4 h after their release from the fuel without any deposition. Table 15.12-6 summarizes the fission product release fractions for this transient."

15.12-10 and 11; "Radiological Analysis; …Since the large break LOCA [loss-of-coolant accident] event resulted in rapid and total core melt and consequences of the release of the fission product inventory analyzed consistent with Reg. 1.4 guidance, it was chosen as the appropriate bounding case." … "The radiological limits of 10 CFR 100 (25 rem whole body, 300 rem thyroid) are assumed to be applicable to **both off-site personnel** and the evacuating personnel on-site."

The above statements are grossly misleading for the following reasons;

1.) The controlling updated EPA Title 40 Protection of Environment (40 CFR 61.92 Standard) states: "Emissions of radionuclides to the ambient air from Department of Energy facilities shall not exceed those amounts that would cause any member of the public to receive in any year an effective dose equivalent (EDE) of **10 mrem/yr** [0.01 rem/yr]."

2.) EPA Standards for Uranium Fuel Cycle Normal Operations to general public, 40 CFR 190.10 state: "The annual dose equivalent does not exceed **25 milli-rems** [0.025 rem] to the whole body, 75 milli-rems [0.075 rem] to the thyroid, and 25 milli-rems [0.025 rem] to any other organ of any member of the public as the result of planned discharges of radioactive materials, radon and it daughters excepted, to the general environment from uranium fuel cycle operations and to radiation from these operations." [12]

3. NRC 10 CFR 100.11 states in part; "However, **neither its use** … as set forth in these site criteria guides are intended to imply that these numbers constitute acceptable limits for emergency doses to the **public** under accident conditions." Additionally this NRC guidance is for a "**once in a lifetime** accidental dose for radiation workers…" The fact is ATR/RTC personnel receive regular radiation doses that must necessarily be factored into any additional accident doses. Therefore, DOE's attempt to use NRC guidance in "off-site" exposures is false. See Section III below for more details on radiation standards. [Emphasis added]

15.12-13; "Confinement Release Rate; The total source term at 100%/day leak is assumed displaced [released to the environment] in 24 hours."

Since the ATR is housed in a ordinary steel sheathed industrial building built in the 1960s, there is no credible containment. Therefore, DOE's claim to 24 hour source term is not credible because it would be near immediate – especially during an accident.

DOE's own 2007 Occurrence Report states; "The ATR Design Basis Reconstruction Project identified five issues with the ATR safety basis evaluation of potential confinement over-pressurization as follows;

1.) The Safety Analysis Report over states the capability of the confinement to withstand an over-pressure event.

2.) The Remote Monitoring System functions for confinement over-pressure protection was eliminated without adequate evaluation.

3.) Confinement leak performance data has been extrapolated far beyond the range of measured data.

4.) The SAR does not adequately account for potential confinement heat sources.

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[12] See Institute for Energy and Environmental Research, Science for Democratic Action, August 2009, for detailed critical analysis of the inadequacy of EPA, DOE and NRC exposure regulations to the public. www.ieer.org

Confinement under-pressure events have not been evaluated." [NE-ID-BEA-ATR-2007-0022]

15.12-14; "Exposure for Evacuating INEEL [sic] workers….the resident time for TRA workers is 1200 s (2 min for activation of the evacuation alarm + 18 minutes to load busses.."

This above statement is not legitimate because the worker buses always return to their base at Central Facilities Area (CFA) several miles away for servicing and do not return until shift-change. Depending on the timing of the driver and TRA shift change considerable time could elapse between mobilizing off-shift drivers and the time to get buses from CFA to the TRA/ATR for worker evacuation. Therefore, the worker exposure time could be significantly longer.

15.12-17; "The relocation of molten [reactor fuel] core debris to the flow distribution tank was performed with user-defined slumping…during a [loss-of-coolant accident] LOCA…that was not capable of simulating the melt and relocation of ATR fuel elements in a manner consistent with the LOCA assessment. There are limitations in this approach…" "The initial temperature of the relocating material [molten ATR fuel debris] was assumed to be 1250 K [Kelvin] [976.84 Celsius], a conservative estimate for melt held up in the core a few seconds after melting within 3-5 seconds of scram." "It is believed that some fission products evolve from the fuel when the Al [aluminum] fuel when the Al cladding melts, exposing the U-AL [uranium-aluminum] fuel matrix. Additionally, experimental data shows that fission gasses evolve from U-Al fuel starting at the melting temperature of AL (922 K)." 922 Kelvin = 648.84 Celsius. [13] "Additionally, as relocated [ATR molten fuel] material heats up in the tank, more fission products are likely to evolve. For these reasons, it is decided to determine at what power level the flow distribution tank might be expected to survive without structural failure."

15.12-18; This was **redacted** by DOE with no credible explanation. It must be noted that the INL press release 11/17/09 states: "The Advanced Gas Reactor (AGR) Fuel Program, initiated by the Department of Energy in 2002, used INL's unique Advanced Test Reactor (ATR) in a nearly three-year experiment to subject more than 300,000 nuclear fuel particles to an intense neutron field and temperatures around 1,250 degrees Celsius."

This fuel temperature exposure of 1,250 Celsius compared to the above 648 Celsius acknowledged melting point for ATR fuel = 602 degree difference which explains why there are dozens of ATR fuel failures/reactor scrams (two so far in 2009). [14]

The major safety problem of ATR fuel failure is acknowledged in DOE's "Potential Inadequacy in the Safety Analysis (PISA), Impact for Potential Leakage on Fuel Element Thermal-Hydraulic Conditions Prior to Reactor Vessel Venting" (RTC-USQ-2005-173), approved 4/4/05. [15] This report challenges the adequacy of the earlier 2004 UFUFSAR as "non-conservative" and states: "The safety basis does not include analysis of a complete loss of coolant in the [Primary Coolant System] PCS leakage. The event analyzed assumes the PCS has

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[13] The conversion from temperature data from Kelvin (K) to Celsius (C) is subtract Kelvin by 273.16 to get Celsius.
[14] DOE documents show 13 ATR scrams between 1991 and 2009.
[15] This report was gained by EDI in an unrelated DOE/INL FOIA request in 2007.

no leakage and thus pressure remains high throughout the event. Successful long-term mitigation of the safety basis event requires activation of the manual vessel vent valves so that the emergency firewater injection system can add water to prevent fuel damage….The emergency procedure development also did Not recognize the potential for PCS leakage to reduce pressure prior to actuation of the vessel vent valves…. A scoping calculation shows severe fuel clad temperature excursions could occur prior to reaching the elapsed time for vessel vent valve actuation if the PCS pressure decreases."

Page 2 of this report was redacted by DOE.

During startup of the Advanced Test Reactor on March 8, 2009, it was determined that a primary coolant check valve was not seating properly. Startup preparations were stopped, the primary coolant system was depressurized and the reactor was defueled so the check valve could be replaced. (NE-ID-BEA-ATR-2009-0003).

However, as noted below [15.15], the power peaking limit must be reduced or the calculated power perking must be increased in the evaluation process (anticipates the reflector crack occurring during the operation)….Therefore, the inner plate limits are reduce by 2.5% to 406 MW for two-pump operation and 431 MW for three-pump operation."

The above ATR power level of 431 MW is nearly twice the DOE ATR power level claim of 250 MW that was further reduce to 150 MW due to unresolved safety problems. [See Facility Certification Report No. 29 by Battelle Energy Alliance catalogues equipment failures and malfunctions due to age of the ATR and unavailability of replacement parts.]

This increased power level exacerbates the hazards related to fuel failure. "Could the PISA increase the consequences of an accident previously evaluated in the safety basis? Yes. The ATR UFSAR Sections 15.3.3 and 15.6.7 evaluate a very small seismic LOCA [loss-of-coolant accident] and a CLOFA [complete loss-of-coolant accident] to both meet the plant protection criteria. PCS [primary coolant system] leakage such that the vessel pressure is just above EFIS [emergency fire water injection system] actuation pressure when battery-backed ECP [emergency coolant pump] flow is lost was not previously evaluated. Because, the consequences of these two accident sequences might now lead to fuel damage, the consequences of an accident previously evaluated in the safety basis increases."

On page 4, the above report states: "Could the PISA reduce a margin of safety as defined in the safety basis? Yes…..Since the current method of determining the time interval to reach 200 degree F. might be much longer than 30 minutes (depending on the plant's condition), the margin of safety defined in the safety basis is reduced."

Page 5 of the above report states: "Does the PISA constitute an Un-reviewed Safety Question? Yes….The potential increase in consequences of an accident previously evaluated and a reduction in the margin of safety indicates that a USQ [un-answered safety question] exists."

15.12-19; In the event of an ATR LOCA accident where molten fuel "relocates" to the bottom of the reactor DOE analysis shows that: "By 415 s [seconds] 168.45 kg of metallic debris had oxidized, producing 18.7 kg of hydrogen. The hydrogen generation rate at 415 s was 0.64

kg/s." It is crucial to note that these events occur nearly immediately (in about a minute) and produce a significant hazard of a hydrogen explosion. This is due in part to "boil-off" of coolant water (also a potential steam explosion) where the coolant "water could not flow and act as a moderator." See Page 15.13-1 below.

15.12-22 through 23, Table 15.12-7; ATR inventory at scram is 1.11 E+9 or **1,110,000,000** curies and the source term 1.74E+08 or **174,000,000** curies that includes **Iodine species source terms at 4.25E+6 or 4,250,000 curies.**

15.12-24, Table 15.12-10; "Potential doses (rem) for a large LOCA in the ATR…Exposed to plume passage at the outer edge of the LPZ (that includes the public on State highways 20 and 23) for Thyroid, 185 rem (185,000 mrem) and TEDE 13.2 rem (13,200 mrem)." Compare these doses to the EPA regulatory limit of 10 mrem/yr cited above.

15.13-1; "High pressure can occur in the PCS [primary coolant system] as a consequence of failure of components or as a result of miss-operation." Four vulnerabilities are identified;

"A. A primary pump discharge valve or check valve closes abruptly;

B. An emergency coolant pump discharge check valve is open with the PCP's [primary coolant pumps] in operation;

C. Startup of the PCP against a fully closed discharge valve with suction pressure significantly higher than 230 psig;

D. The FCV (butterfly) fails."

"These events result in a dynamic pressure pulse resulting from water hammer. The check valve will have the more serious consequence."

UFSAR 15.13-2; "A PCP [primary coolant pump] check valve failed to close properly during a trip of one out of three primary pumps on December 11, 1967. Analysis of this fault showed the maximum calculated pressure was 900 psig….The design pressure is 567 psig. Some pipe hangers were damaged….A repeat of the event should be avoided."

This 1967 accident occurred when the ATR was in its first years of operation. Now 42 years later – long past the ATR design life of 20 years - after the effects of radiation/corrosion deterioration, the primary coolant system could not be expected to survive a similar accident; especially given the documentation that ATR operating contractor is using non-certified replacement parts. [See below] DOE considers these valve failures "low-probability" events and arbitrarily categorizes them in a lower hazard (condition 3) status. This is has no credibility realistically or statistically. When a failure has already occurred, statistically it moves way up the probabilistic scale because it is no longer a hypothetical event.

The critical safety problem of ATR contractors using non-certified replacement parts is acknowledged in a DOE document that states; "The revision to this CGI [Commercial Grade Item] is to **eliminate** the 'performance characteristic' item of the CGI dedication plan as requirement for staging clutch plates in the warehouse. Initially this material was for direct installation. Now, material is to be staged in the warehouse for future use without providing proof of 'performance characteristics', however, proof of 'performance characteristics' will be provided upon installation of any new clutch plates." [16] [Emphasis added] In essence, DOE has suspended its previous "commercial grade" requirement for the most crucial mechanisms for primary coolant system and emergency shutdown of the ATR.

15.15-1; Section 15.15 Reflector Aging; "The ATR reflector is a machined beryllium structure located in regions of significant gradients in the neutron flux. The resulting non-uniform exposures result in non-uniform growth and internal stresses. The internal stresses eventually result in cracking of the thin ligaments and spalling [sic] along he cracks….The ligament cracking can result in replacement of beryllium with water. This replacement increases the power peaking in the adjacent fuel element….The analyses for fuel elements adjacent to reflector ligaments that are cracked (or may be near cracking) established an acceptable (reduced) lobe power for the standard 7F fuel element….The effective plate powers for the inner plates remain the same as for the new reflector (417 MW for two-pump operation and 443 MW for three-pump operation….However, as noted above, the power peaking limit must be reduced or the calculated power perking must be increased in the evaluation process (anticipates the reflector crack occurring during the operation)….Therefore, the inner plate limits are reduce by 2.5% to 406 MW for two-pump operation and 431 MW for three-pump operation."

The above ATR power level of 431 MW is nearly twice the DOE ATR power level claim of 250 MW that was further reduce to 150 MW due to unresolved safety problems. [See Facility Certification Report No. 29 by Battelle Energy Alliance catalogues equipment failures and malfunctions due to age of the ATR and unavailability of replacement parts.]

15.16-1; "Results of the PRA [preliminary risk assessment] …Of the total fuel damage frequency for external initiated event, seismic (reactor core fuel damage) events contributed 84% …the event has a site-independent frequency that is not negligible (fuel damage frequency was greater than 1.0E-7/yr)."

15.16-2 Section 15.16.1 Internal Fire; "These fire-induced CLOCAs [complete loss-of-coolant accident] in the pump motor corridor or switch gear dominate the previous fire PRA [preliminary risk assessment] results because the new accident analysis success criteria in the UFUFSAR (Section.0.14)" The following paragraph is redacted by DOE. **WHY**???

15.16-3; "ATR core damage may occur if a chemical release results in an explosion the damages the safe shutdown equipment at ATR or a release results in toxic vapors that cause the control room to become uninhabitable." The next paragraph is redacted! WHY?

[16] Commercial Grade Idem Dedication Plan, CGI-303 Revision 1, 2/28/05.

Section III. Emergency Management Hazards Assessment for Reactor Technology Complex (RTC) formerly called Test Reactor Area (TRA) that includes Advanced Test Reactor, Rev.7, HAD-3, 2004

Identified below as HAD-3 Court Printout sections followed by document ascending page number;

Document text excerpts are in "quotation marks" and **Bold** only emphasizes original text font.

Crucial information needed to assess the information below comments

Radiation dose unites used in DOE documents cited below include:

Rem = When relative biological effectiveness of a particular radiation type (relative to gamma radiation or alpha radiation is taken into account, the value is multiplied by a quality factor to yield the units rem. The current rem and the older rad unit are equivalent.

TEDE = Total Effective Dose Equivalent; or whole body that includes "internal" (dose received by a radiation source inside the body, e.g. an inhaled dust particle containing plutonium or ingested contaminated water); AND "external" (dose received by a radiation source exposure outside the body, e.g. from a gamma/alpha emitting radionuclides in soil or air).

CDE = Committed Effective Dose Equivalent; The dose value obtained by (1) multiplying the committed dose equivalents for the organs or tissues that are irradiated and the weighting factors applicable to those organs or tissues, and (2) summing all the resulting products. Committed effective dose equivalent is expressed in units of rem or sieverts.

Source Terms = Radiation released from a specific source or combined with related sources during a specific defined time frame.

Radiation Standards

Updated EPA Title 40 Protection of Environment (40 CFR 61.92 Standard) states: "Emissions of radionuclides to the ambient air from Department of Energy facilities shall not exceed those amounts that would cause any member of the public to receive in any year an effective dose equivalent (EDE) of **10 mrem/yr** [0.01 rem/yr]."

EPA Standards for Uranium Fuel Cycle Normal Operations to general public, 40 CFR 190.10 state: "The annual dose equivalent does not exceed 25 milli-rems [0.025 rem] to the whole body, 75 milli-rems [0.075 rem] to the thyroid, and 25 milli-rems [0.025 rem] to any other organ of any member of the public as the result of planned discharges of radioactive materials, radon and it daughters excepted, to the general environment from uranium fuel cycle operations and to radiation from these operations." [17]

The above 25 mrem for the whole body, critical organ and 75 mrem for the thyroid in 40 CFR part 190.10 as cited above is for all pathways. That is drinking water, air, ingestion of

[17] See Institute for Energy and Environmental Research, Science for Democratic Action, August 2009, for detailed critical analysis of the inadequacy of EPA, DOE and NRC exposure regulations to the public. www.ieer.org

contaminated food, etc. The above (40 CFR 61.92 Standard) 10 mrem is for emission of ambient air - applies only to one pathway. However, **both standards** have to be met separately.

Although the Nuclear Regulatory Commission's "Reactor Site Criteria" sets lower exposure standards, it states; "However, neither its use … as set forth in these site criteria guides are intended to imply that these numbers constitute acceptable limits for emergency doses to the public under accident conditions." [10 CFR-100.11]

HAD-3 [Court Printout 54-15]

Doc. Pg. 5 through 19, seven pages (including 2 pages of acronyms) are **Redacted** ! DOE's extensive use of acronyms makes understanding the information without this data extremely difficult because the definitions are rarely in the text. This is clearly a deliberate attempt to frustrate any independent analysis of these reports.

Doc. Pg. 21 and 22, "Hazardous Materials Screening Results," are **Redacted**.

Doc. Pg. 32 and 34,Table 6, "TRA-670 (ATR and ATRC) Radiological release scenarios that have the potential for being classified as operational emergencies"); "ATR/ATRC Accident Fission Product Release Downwind Dose (in Rems at 30 meters, 2.5 hours) Scenario;

TEDE 7.38 E+6; 7,380,000 rem

CDE 9.00 E+7; 90,000,000 rem

Rem/Hour; 516 R/h" (See discussion below)

Doc. Page 42; "ATR Primary Coolant Treatment at TRA-605, Table 10 ;

Thyroid TEDE 1.54 E+4; 15,400 rem

Thyroid CDE 3.04 E+5; 304,000 rem"

HAD-3 [Court Printout 54-16]

Doc. Page 54; "Consider relative probability of the worst-case scenario" **Is Redacted-Why ?** **"**Analysis: Two of the non-radiological release scenarios exceed the PAC beyond RTC's existing 5-km EPZ [emergency planning zone] radius. The maximum PAC distance for non-radiological scenarios is 7.8 km."

Doc. Pages 57 through 99 (42 pages) are **Redacted.**

Doc. Page 102 through 113 (11 pages) "ATR Facility and Process Description" are **Redacted.**

Doc. pg. 114; ATR Canal; "The inventory of concern is related to the fission products contained in the fuel elements from the most recent refueling operation. The fission product and activation product inventory at the reactor scram following 60 days of continuous operations at 250 MW is 1.11 E+9 Ci [[1,110,000,000 Ci].

Table A.1-3 presents the inventory of radionuclides that have the most significant contributions to radiation dose. This inventory (8.91 E+7 Ci) [89,100,000 Ci] is used as the starting basis for all accident scenarios that are evaluated in this document."

Doc. Pages 116 to 128; "ATR Plant Damage States" are **redacted.** WHY ?

Doc. Page 129; "Source Term: 5.0 E+6 **R/h** at 1 ft." [**5,000,000 Rem/<u>hour</u>**] "RTC [where

ATR is located] main parking lot at 500 m could be approximately 2 R/h. Table A.1-20 [later on page 135] provides exposure rate information at several distances, which are based on an initial exposure rate of 5.0 E+6 R/h at 1 foot (0.3048 m) and applying the inverse square equation. Other Barriers and Effects: Although the ATR building structural materials may provide some shielding, no credit is taken for shielding." [Because none can be legitimately be claimed]

Doc. Page 130; "ATR Small Break LOCA [loss-of-coolant accident], Core Melt Source Terms, Table A.1-10; All Iodine species 5.82E+5 Ci, [582,000 Ci]"
Total Krypton, Xenon, Cesium, Tellurium, and Iodine, 2,617,998 Ci.

Doc. Page 132; "Table A.1-13, ATR very large reactivity insertion accident (RIA) core melt source term; Iodine species 5.82 E+5 [582,000 Ci.];
Total all isotopes 2,617,998 Ci."

[RIA - Reactivity Insertion Accident; "A perched fuel element drops into core after criticality is achieved, which causes a very large and rapid reactivity insertion that causes vessel movement and rupture of piping leading to loss-of-coolant" (LOC)]

Doc. Page 133; Table A.1-16; ATR Canal Drain Core Melt Source Terms, 90,046,100 Ci that includes all iodine species 5.70 E+7 [57,000,000] Ci

Doc. Page 135; "Table A.1-20. ATR dropped loop experiment exposure rates by distance and ATR dropped core internal change-out irradiated parts or equipment [lists];

30 meters [98.4 feet]………. **516 R/hr**
68 meters [223 feet] ……… **100 R/hr**
485 meters [1,590 feet]……… **1.97 R/hr"**

[**The above table shows that all personnel at the ATR would receive a lethal radiation dose and depending on the evacuation time all the RTC personnel would receive life-threatening doses. Since most INL personnel are transported (to and from work) by DOE's fleet of buses (based at Central Facilities Area several miles away and only go out during shift changes), evacuation time could be significant – thus increasing exposure doses.]**

Doc. Page; 139; Table A.1-22, ATR LOC, Downwind TEDE at 30 meters, in Rems
four 4 hr 4.25E+4 [42,500];
forty-eight hr. 5.10E+5 [51,000]

Doc. Page; 142; Table A.1-22, ATR Canal Drain Downwind TEDE LOC, 2 hrs in Rems;
7.38E+6 [7,380,000]

Doc. Page; 146; Table A.1-23, ATR LOC, Downwind CDE (thyroid) at 30 m., in Rems
at 4 hr. 2.66E+5 [266,000];
at 48 hr. 3.19E+6 [3,190,000]

Doc. Page; 148; Table A.1-23, ATR LOC, Downwind CDE (thyroid) at 30 m., in Rems
at 4 hrs, 6.69E+5 [669,000] ;
at 48 hrs. 8.03E+6 [8,030,000];
At INL boundary at 4hrs, 2.69E+1 [26.9]
at 48 hrs. 3.23E+2 [323.0]

HAD-3 [Court Printout, 54-19]

Doc. pg. 295 to 355; Covers ATR Storage Facility (TRA-634 Hot Side). **Redactions** of three pgs. 327 to 329.

Doc. Pg. 295; Table A-4-12; ATR "RTCHC test train and isotope production run exposure rates by distance (30 meters) in **Rem/hour**; Test Train 516 R/h
Isotope Production Run 35.6 R/h

It is uncertain if these exposure rates include ATR core exposure rates or are in addition to them. (See above discussion of Doc. Page 32 and 34).

Doc. Pg. 321; TRA-634 has an inventory of mixed fission products of 1.75E+5 (175,000) Ci This printout also includes ATR Cooling Tower Pump House (TRA-671) "[T]hat is one of the facilities that is necessary for safe operation of the ATR. TRA-671 contains the secondary cooling pumps to maintain proper cooling of the ATR as well as water treatment equipment and water treatment chemical storage tanks." [pg. 334] A major earthquake or large fire must be considered in total potential radiation releases.

HAD-3 [Court Printout. 54-20]

Doc. pgs. 356 to 410] that include ATR Primary Coolant Systems (PCS) in TRA-605. "TRA-605 is part of the Effluent Processing EP) Facility whose mission is to collect, store, process and dispose of radioactive contaminated liquid waste streams generated at the Advanced Test Reactor (ATR)." [pg. 365]

It also states; "It is recognized that any release from TRA-605 would be due to an event that has already occurred at the ATR. The release from TRA-606 is a secondary event that could occur during recovery operations to mitigate the original event." [pg 374]

a. ATR (TRA-670 **and** TRA-605) Core (only/no experiments) (Table A-7.1; pg.371);
Inventory is 519,129,000 Ci

b. ATR Core (only no experiments) [Table A-7.2; pg.372]
8.91E+07 = 89,100,000 Ci;

c. Primary Coolant System (PCS) TRA-605; Source Terms (100% fuel melt) Table A-7.6; pg. 381; 1.25E+08 = 125,000,000 Ci..

d. Primary Coolant System (PCS) TRA-605; Source Terms (100% fuel melt) Table A-7.9; pg. 382; **Iodine species; 1.38E+7** **13,800,000 Ci..**

c. Primary Coolant System (PCS) TRA-605; Radiological Release scenario Total Effective Dose Equivalent (TEDE) also known as whole body dose; **Minor** Fuel Cladding Damage Causes PCS; Table A-7.13;
Collocated Worker 1.54E+4 =15,400 rem;

INEL Boundary (public) 5.63E-1 rem = 563 mrem.

d. Primary Coolant System (PCS) TRA-605; Table A-7.13; Downwind **Thyroid** Committed Dose Equivalent (CDE); **Minor** Fuel Cladding Damage;
Collocated Worker 3.04E+5 rem = 304,000 rem;
INEL Boundary (public) 1.20E+1 rem = 12.0 rem = 12,000 mrem.

These above HAD-3 sections contain doses and source terms for ATR, Canal, and PCS "treatment" facility failures. It is uncertain if these releases; 1.) are in addition to or included in the other ATR summary accident doses/source terms identified in (HAD-3); AND/OR 2.) Are these in addition to ATR experimental fuel inventories and the respective releases?

The frequency of ATR experimental fuel failures that release fission products into the Primary Coolant System which then requires frequent change-out of coolant water that is sent to TRA-606 for "treatment." Thus the high radiation releases from TRA-605. Since the heat exchangers between the primary and secondary coolant systems, the releases from the ATR coolant towers are also connected to fuel failures.

Section IV. Engineering Design File 4394, Update of ATR Break Spectrum and Direct Loss-of-Coolant Accident LOCA Frequency Analysis released by order of U.S. Federal District Court, District of Wyoming.

Court Document Printout 54-25, Filed 10/13/09, Exhibit 4 Declaration of Karl J. Hugo, Redacted Version of the Engineering Design File 4394 Update of ATR Break Spectrum and Direct Damage Loss of Coolant Accident Frequency Analysis.

Identified below by document ascending page number (EDF-*);

Document text excerpts are in “quotation marks” and **Bold** only emphasizes original text font.

EDF-1; “Summary: One of the key assumptions in the ATR UFSAR is that direct damage loss-of –coolant accidents (DDLOCAs) are beyond design basis events. The analysis supporting this finding relies on 1) break spectrum analyses that identified break sizes and location that would lead to direct damage at various lobe maximum power levels, and 2) restricting the operating time at high lobe power levels to keep the **frequency of a potential DDLOCA acceptably low.”**

“Changes in the ATR systems and operation, specifically, the emergency firewater injection system (EFIS) pressure set-point and the addition of the LOCA primary pump shutoff engineered safety feature, have been made since the last break spectrum analyses were performed. The impact of these changes on the break spectrum analyses needs to be evaluated to ensure that the current UDSAR assumptions regarding piping susceptibility to a DDLOCA are still valid.”

The remainder of this Summary is **Redacted** ! Why ? DOE’s above statement “restricting the operating time at high lobe power to keep the frequency of a potential [accident] acceptably low,” is not reassuring to the downwind public given the magnitude of **any** significant ATR accident for discharging huge amounts of radiation into the environment.

EDF-12; Except for the first two lines, the rest of the page is **Redacted** ! Why?
EDF-13; “Inlet Pipe Break;” Two-thirds of the text is **Redacted** ! Why ?
EDF-16; Most of this page is **Redacted** ! Why ?
EDF-17; Under “36-in.Inlet Tee Break,” most of this page is **Redacted** ! Why ?
EDF-18; This page is completely **Redacted** ! Why ?
EDF-19; This page is completely **Redacted !** Why ?
EDF-20; This page is completely **Redacted** ! Why ?
EDF-23; “Calculations of 15 large break LOCA in the ATR…” This page is nearly all **Redacted !** Why ?

Section V. Other DOE Divisions Consider Safety Analysis for the ATR 2004 UFSAR is Inadequate

Office of Facility Safety (EH-2) Office of Environment, Safety and Health **Un-reviewed Safety Question Activity Report** July – September 2005 (page 32 [B-1]) reports that higher radiological consequences could result for an accident at ATR than analyzed in the UFSAR-153 report because of a faulty analysis of flow rate in the hot fuel plate analysis. Thus there is a "potentially inadequate safety analysis" for the ATR.

March 2004 Idaho National Engineering Lab/Advanced Test Reactor NE-ID--BBWI-ATR-2004-0004 Core Feedback Failure During Loss of Commercial Power Update issued 08/18/2005

Occurrence Report No. 13, USQ (Un-answered Safety Questions) No. RTC-USQ-2005-336, Discovered: June 15, 2005, and 1610: states; "The ATR SINDA-SAMPLE code models the variation in flow rate in the hot fuel plate analysis. The model development did **not** explicitly address some pertinent sources of uncertainty and therefore may not be conservative."

Occurrence Report No. 14, USQ No.: RTC-USQ-2005-248, Discovered: May 4, 2005, 1630: "The derivation of the analytical limit set-point and response time are not consistent with the methods used in the radiological consequence analyses presented in UFSAR-153, Section 15.7 and 15.12. The methodology used for the derivation of the set-point could allow higher off-site doses than predicted by the radiological consequence analyses. Since these radiological consequence analyses are the basis upon which DOE approved operation of the ATR, the discrepancy represents a potentially inadequate safety analysis." [18]

Operations Report 12/12/07

Nov. 15: During a planned power outage at the Reactor Technology Complex, power was unexpectedly lost to another building in the area. Work in progress, including crane operations and containment work requiring filtered air movers, was impacted. Upon discovery of the unexpected power loss, a decision was made to complete the work in order to restore power quickly to the affected building. A critique was held to determine the cause of the incident and to identify lessons learned. (NE-ID-BEA-ATR-2007-0025).

Operations Report 12/8/09

Nov. 24: Start-up of the Advanced Test Reactor was interrupted by an instrument problem. The problem was diagnosed and corrected and reactor start-up resumed. (NE-ID—BEA-ATR-2009-0024).

Dec. 2: The Advanced Test Reactor was shut down when a calculation error was discovered in the assurance package for that particular reactor operating cycle. The reactor remained in shutdown until the error was corrected and a re-calculation performed. (NE-ID—BEA-ATR-2009-0025).

[18] http://www.eh.doe.gov/facility/safety/usq-activity-report-205-3.pdf

Operations Report 1/21/10

Jan. 12: The shift supervisor at the Advanced Test Reactor entered into a limiting condition for operation of the reactor when two instrument systems used to calculate water flow in the reactor were declared out of service. Limiting conditions for operation are a Department of Energy approved method to ensure safety of nuclear facilities while system performance is evaluated. The shift supervisor used other data systems to verify the safety of reactor operations while the systems were repaired and returned to operation. (NE-ID—BEA-ATR-2010-0001).

Section V. Other DOE Documents on ATR Operations

One of the missions of the Advanced Test Reactor (ATR) is to irradiate new reactor fuels – in support of the DOE Advance Gas Reactor Fuel Development and Qualification Program. This is the first of eight similar test experiments to test the fuel cladding's ability to provide a barrier to fission product release in a high temperature (1,250 degrees Celsius) and high radiation condition. Inside the ATR reactor, the fuel specimens were subjected to neutron irradiation many times higher than what they would experience inside a High-temperature Gas Reactors. A recent fuel test program called AGR-1 on "uranium oxi-carbide in a graphite matrix" found that the fuel coatings failed - releasing radio-iodides and cesium into the reactor and released to atmosphere.

DOE conducted a 12/06 interim analysis of radioactive Iodine-135 releases after the first six week fuel testing program ended when the ATR was "scrammed" (emergency shutdown) likely due to fuel failure. According to DOE's report, this program continued for an additional 30 weeks (2.5 yrs) to June 2009. It must be noted that numerous species of longer-lived radioactive iodine in addition to I-135 (1-129, I-131, I-132, I-133, and I-134) as well as many other radionuclide fission products (i.e., Krypton, Xenon, Cesium and Tellurium) were also released in significant quantities. [19]

Specifically, the AGR-1 report acknowledges release of 1,200,000 pico-curies of I-135 over the 6 week monitoring. As the report also acknowledges the program lasted 30 weeks; the estimated I-135 releases are about 2,327,400 pico-curies. [20]

This is a significant release that EDI believes violates radiation emission standards as the discussion below documents. Pico-curie (one trillionth of one curie) units are used here because they are used by regulatory agencies due to their extreme biological hazard to human health. ." **If EPA and State of Idaho regulatory limits on radiation exposure to the public from both DOE and commercial nuclear power reactors, it indicates violations. [21]**

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[19] Emergency Management Hazards Assessments for Reactor Technology Complex, 2004, HAD-3 page 132. This report acknowledges release of 2,617,999 Curies of these isotopes released in ATR fuel failure accident.

[20] "Determination of the Quantity of Iodine-135 Released from the AGR-1 Test Fuels at the End of ATR Operating Cycle 138B," J.K Hartwell et...al., May 2007, Idaho National Laboratory, INL/EXT-07-12455. On page 11, the report states "At the conclusion of the ATR operating cycle 138B the activity of released I-135 for **each** [of the six] test fuel capsule was determined to be less than 0.2 micro-curies." For
the six capsules the release = 1.2 micro-curies or 1,200,000 pico-curies. Over the total 30 week program (through 6/09) the estimated I-135 release is 2,327,400 pico-curies.

[21] Environmental Protection Agency Radiation Protection Standards for the Environment (40 CFR-61.92) states: "Emissions or radionuclides to the ambient air from Department of Energy facilities shall not exceed those amounts that would cause any member of the public to receive in any year an effective dose equivalent of 10 mrem/yr."

In a related document, an internal ATR report, DOE's analysis states the following radiation releases; "Condition 1 **(Normal operation)** – Radiation exposure of: 1.00 mSv/year [100 mrem/yr or 10 times over regulatory limits] effective dose equivalent (EDE) and 0.10 mSv/year [10 mrem] EDE from airborne release to off-site public and 0.05 Sv/year [5,000 mrem] total effective dose equivalent (TEDE) to workers. Reactor fuel source term protection limit: The integrity of the reactor fuel cladding is NOT challenged except for limited clad defects."

"Condition 4 [**fuel failure as described above in detail]** – Radiation exposure of: 0.25 Sv [25,000 mrem or 2,500 times over regulatory limits] whole body and 3.00 Sv [300,000 mrem or 30,000 over regulatory limit] thyroid dose to off-site public and evacuating workers (excluding personnel considered directly at the location of the accident). Reactor fuel source term limit: The reactor primary coolant pressure boundary must be maintained (unless this failure is the initiator) and the reactor confinement must not be damaged. The predominant risk associated with the ATR is the radiological source term [release to the environment] contained within the reactor fuel." [22]

The public can be justifiably outraged that the ATR is exempt from radioactive emission compliance required of commercial nuclear power reactors when clearly the ATR releases shown below are in violation of EPA regulatory standards; however, even DOE's Order (5400.5) that limit radiation emissions to 100 mrem whole body to the public is also violated. [23]

Review of Determination of the Quantity of I-135 Released from ATR

A quick glance of this report, it looks relatively unimportant. EDI's review however shows more revealing information. DOE is using radiation measurement units that are a million times lower (nano-curies) than the regulatory standards of pico-curies. Therefore, a casual reader may overlook the importance of the report.

On page 11 Conclusion the report states that each of the six test fuel capsules released 0.2 micro-curies. This = for all six capsules 1.2 E -6 micro-curies that equals 1,200,000 E-12 pico-curies that are the units used in regulations due to the significant health effects of radionuclides.

This report only shows a 6 week monitoring of a 2.5 year (30 week) program (12/25/06 to 6/09) as stated on page iii.

Assuming, as the reports stated 30 week program, it is possible that 2,327,400 pico-curies of I-135 were released from the ATR during the 30 week program. It can be assumed that numerous species of radioactive iodine with longer half-lives (i.e. I-133 and I-129) as well as many other long-lived radionuclides were also released. See Table below.

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[22] "Safety Assurance for Irradiating Experiments in the Advanced Test Reactor," T.A.Tombererlin, at.al. Idaho National Laboratory, 11/14/04, INEEL/CON-04-02244. Severt (Sv) is the international unit of radiation exposure (one Severt = 100 rem); Rem is a unit of radiation exposure used in the US. 1.0 mSv = 100 mrem.

[23] Tragically, thanks to Congress, the EPA and the State of Idaho has no jurisdiction over the ATR. DOE Radiation Protection of the Public and the Environment; Order 5400.5 (1/7/93) however is violated.

ATR Experimental Fuel Failure Releases

ATR Operating Condition	Total/Yr Radiation Exposure	EPA /Yr. Regulatory Limit *	Air Release Off-Site	EPA / Yr. Regulatory Limit Fuel Cycle **	Worker Exposure / Yr.	DOE/Yr. Regulatory Limit to Public [24]
Condition 1 Normal	100 mRem	10 mRem	10 mRem	25 mRem	5000 mRem	100 mRem
Condition 2 Anticipated Faults	N/A	10 mRem	500 mRem	25 mRem	5000 mRem	100 mRem
Condition 3 Unlikely Faults	6250 mRem	10 mRem	750000 mRem Thyroid	75 mRem	N/A	100 mRem
Condition 4 Extremely Unlikely	25000 mRem	10 mRem	300000 mRem Thyroid	75 mRem	N/A	100 mRem

* Updated EPA Title 40 Protection of Environment (40 CFR 61.92 Standard) states: "Emissions of radionuclides to the ambient air from Department of Energy facilities shall not exceed those amounts that would cause any member of the public to receive in any year an effective dose equivalent (EDE) of **10 mrem/yr** [0.01 rem/yr]."
** EPA Standards for Uranium Fuel Cycle Normal Operations to general public, 40 CFR 190.10 state: "The annual dose equivalent does not exceed 25 milli-rems [0.025 rem] to the whole body, 75 milli-rems [0.075 rem] to the thyroid, and 25 milli-rems [0.025 rem] to any other organ of any member of the public as the result of planned discharges of radioactive materials, radon and it daughters excepted, to the general environment from uranium fuel cycle operations and to radiation from these operations." These standards are presented for comparison only because: "Certain releases of radionuclides may qualify as a 'federally permitted release.' A release of source, special nuclear, or byproduct material is a federally permitted release if such release occurred in compliance with a legally enforceable license, permit, regulation, or order issued pursuant to the AEA [CERCLA §101(10)]. This includes a release in compliance with DOE Orders [see DOE Order 5400.4(4)]."

Although apparently technically accurate, DOE consistently uses units of radiation exposure that makes the data in their reports appear innocuous. EDI offers a translation of the data into radiation units used in federal/state regulations. As DOE's own data shows above, the ATR is apparently in violation of regulatory limits – just for this single of numerous concurrent and ongoing ATR experimental fuel programs.

Tragically, the above discussion of test fuel failure is more normal as opposed to abnormal, based on many DOE internal documents acknowledging fuel failure. [25] DOE steadfastly refuses to release current ATR radiation data.

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[24] DOE Order 5400.5; Radiation Protection of the Public and the Environment (pg. I-2). "DOSE LIMIT SELECTION. The DOE primary standard of 100 mrem (1 mSv) effective dose equivalent to members of the public in a year is lower than the previous primary limit of 500 mrem (5 mSv). The lower value was selected in recognition of the ICRP recommendation to limit the long-term average effective dose equivalent to 100 mrem (1 mSv) per year, or less. Experience suggests that the lower dose is readily achievable for normal operations of DOE facilities. A higher dose limit, not to exceed the 500-mrem effective dose equivalent recommended by ICRP as an occasional annual limit, may be authorized for a limited period if it is justified by unusual operating conditions."

[25] Occurrence Report, After 2003 Redesign, ATR N-16 System Degradation Results in Manual Reactor Shutdown, NE-ID- BEA-ATR-2008-0001, 1/9/08

DOE's own previous Environmental Impact Statement states: the ATR released 1,802 curies in 2000 and 1,180 curies in 2003 to the atmosphere. [26] On average that is about 1,491 curies/year; so over an eight year period 2000 through 2008 (given ATR's continuous operation) about 11,928 curies may have been released to the air. These high emissions from ATR suggest liquid waste is first sent to the ATR cooling towers w/o treatment and the precipitates are then pumped to INTEC evaporators or the percolation ponds. This represents a significant hazard to INL workers and the downwind public.

By any standards, these are significant releases that have a major health impact on the downwind uninformed public! **None of the ATR missions are so crucial to the national interest that it justifies the enormous risk to the public during "normal" operations and/or in the event that this antiquated reactor has a meltdown via system failure and/or an earthquake initiating cascading reactor system failures.**

[26] DOE/EIS-0287 pg. 4-30; DOE/DEIS-0373D, pg 3-26.