SUPPLEMENT NO. 1

to

SAFEGUARDS REPORT
FOR THE SAXTON REACTOR PARTIAL
PLUTONIUM CORE II

May 1965

Question #1 - In order to provide a basis for evaluating the conservatism of the parameters used in the accident evaluation sections of the report, provide verification that the physics parameters measured in the critical experiment at WREC are at least as conservative as those assumed for the accident evaluations. In addition, verify that the proposed loading will be with a central plutonium region.

Answer: The series of critical experiments outlined in the Safeguards Report for the Partial Plutonium Core II is now in progress at the Westinghouse Reactor Evaluation Center (WREC). Although the entire series is not yet completed, the results obtained to date show that experiment and analysis are in excellent agreement and verify that a conservative approach was followed in the design of the Partial Plutonium Core II. While additional experiments and data processing and reduction are continuing, the program is sufficiently complete to be able to state that:

(a) Any data and results obtained in the future are not expected to significantly alter the above conclusions and

(b) The initial core loading will be with the nine plutonium enriched fuel assemblies in the center of the core.

The preliminary results of the criticals which are available are summarized below. The experimental program and series of criticals being conducted at the WREC are outlined in Table 1-1. Predictions as to the number of fuel rods required for criticality, calculated k_{eff} and corresponding boron concentrations are included in this table. The status of the experimental program of Table 1-1 is shown in the following list:

Configuration	Type	Status
A	UO_2-One Region Clean Core	Completed
1	PuO_2-UO_2-One Region Clean Core	Completed except for 1(e)
2	PuO_2-UO_2, UO_2-Two Region Clean Core PuO_2-UO_2 in Inner Region	Completed except for 2 (a)
3	PuO_2-UO_2-One Region Borated Core	Completed
4	PuO_2-UO_2-UO_2-Two Region Borated Core, PuO_2-UO_2 Inner Region	In Progress
5	Two Region, UO_2 Fuel in Inner Region, Clean and Borated	Clean Core-Completed Borated Core-In Progress
6	PuO_2-UO_2-One Region Clean Core, Larger Pitch	To be done

Reactivity Experiment Results

The results of two critical experiments are available for comparison with predicted results. A major portion of the experiments was done with the same H/Pu ratio that will exist in the Saxton reactor at operating temperature (T_{mod}=530°F).

Configuration	Fuel	Pitch	Fuel Rods Req'd for Criticality PDQ Analysis	Experiment
1(c)	PuO_2-UO_2	0.56 in.	355	343
A(3)	UO_2	0.56 in.	356	346

Using the same cross-section data and calculational methods employed in the core design, experimentally determined values of buckling were used to calculate the effective multiplication factors for various lattices and fuels.

Fuel	Configuration	Lattice Pitch	Calculated k_{eff} LEOPARD (Total Buckling)	Calculated k_{eff} X-Y PDQ (Axial Buckling)
UO_2	A(3)	0.560 in.	1.0042	1.0045
UO_2	A(2)	0.792 in.	0.9997	-

			Corrected k_{eff} LEOPARD (Total Buckling)	Corrected k_{eff} X-Y PDQ (Axial Buckling)
PuO_2-UO_2	1(b)	0.560 in.	0.9950	0.9966
PuO_2-UO_2	1(c)	0.792 in.	1.0063	-

For all of these experiments, the experimental k_{eff} was 1.0. Evaluation of k_{eff} for the PuO_2-UO_2 lattices included an allowance of 0.025 which is based on previous comparisons of analysis by these methods with experimental results of a number of Hanford mixed oxide critical experiments so that [Corrected k_{eff} = Calculated k_{eff} - 0.025].

The value of 0.025 was selected prior to completion of the experiment so that its selection was not influenced by prior knowledge of the experimental results of the buckling measurements. The excellent agreement between the analytical predictions and the experimental results shows that the allowance selected was a reasonable one. No allowance was included in the evaluation of the UO_2 results.

From the standpoint of the Saxton core design, the results of the experiments lead to the following conclusions:

(a) There is no need to modify the expected core lifetime or installed reactivity predictions used in the reference design of the Safeguards Report.

(b) The good agreement between analysis and experiment for a wide range of H/Pu ratios indicates that one of the most important factors of the moderator temperature coefficient, the density effect, is correctly calculated by the analytical methods used in the core design.

Power Peaking Results

Power peaking experiments in fuel rods adjacent to water slots have been carried out in both single region and two region cores. Only the results of the single region cores have been analyzed to date. In the single region experiments, a water slot was formed by removing five center fuel rods from a square lattice. The power level in the adjacent fuel rods was measured with and without the water slot. Experiments were also carried out with an aluminum slab in the water slot to displace some of the water. Using the various lattice characteristics, PDQ-3 analyses to predict the peaking effect have been carried out and are compared with experimental measurements.

Peaking Factor Ratio: Analysis/Experiment

Core	H_2O Slot	H_2O + Al Slot
PuO_2-UO_2	1.0779	1.0400
UO_2	1.0555	1.0104

These results demonstrate that the analytical methods used in the Core II evaluation are conservative in that they over-predict the power peaking effects in water slots. These results

are representative of the actual conditions which will be present in Core II as installed in the reactor because the peak in the core occurs within the boundary of the Pu fuel region and is therefore more characteristic of a single region core than peaking at the boundary of a two region core. The results of this analysis demonstrate that the hot channel factors assumed in the core design are conservative and that the initial power level shown in the Core II Safeguards Report may be raised from 21.6 MWt, probably up to 23.5 MWt. Additional testing and low power experiments will determine the actual hot channel factors and initial power level for Core II.

Boron Worth Results

Boron worth measurements were made in the two region core of configuration 4(b). The predicted boron concentration required for a full water height critical was 1525 ppm. The experimental results extrapolated to full water height conditions showed a concentration of 1550 ppm which is in excellent agreement with the prediction.

Kinetic Parameter Results

The kinetic characteristics of single region and two region cores are presently being investigated using pulse neutron techniques. An additional experiment has been completed for a single region Pu core which measured the neutron lifetime by measuring the reactivity change for a small addition of boron (~25 ppm) to the moderator. Although all of the experiments being conducted to determine the kinetic characteristics are not yet complete, these preliminary comparisons of analyses and experiments are available:

Fuel	Lattice	Prompt Neutron Lifetime, ℓ (μ sec)			
		LEOPARD	PDQ (1/v Poison)	Boron Addition	Pulse Neutron
One Region, PuO_2-UO_2	0.56 in.	8.5	19.4	15.8	20.5 (Calculated from β = 0.0034 and measured β/ℓ = 166 sec^{-1}
One Region,	0.56 in.	15.0	20.4	-	30.3 (Calculated from β = 0.00795 and measured β/ℓ = 262 sec^{-1}

As the table shows, the values of ℓ if calculated for the experiment by LEOPARD are much shorter than those inferred from the experiments. This indicates that the actual values of ℓ for Core II will be longer than those predicted by the LEOPARD calculation and reported in the Core II Safeguards Report.

Question #2 - It is proposed that some of the PuO_2 fuel in Core II will operate at specific power levels of up to 16 Kw/ft. To enable us to evaluate any significant safety problems associated with operation at this proposed specific power, provide a discussion of the results of such operation involving UO_2 fuel at the Saxton reactor.

Answer: The peak specific power level of 16 Kw/ft is a conservative design limit based upon present Westinghouse fuel element design practice and techniques. This limit is believed to be a reasonable upper boundary for the initial operation of the mixed oxide, partial plutonium core for Saxton. A great deal of experimental data exists on the successful operation of test fuels of these types (sintered pellets and vibration compacted powder) at specific power levels greatly in excess of 16 Kw/ft and even, in some cases, with significant center melting of the fuel.

The limit of 16 Kw/ft is a reasonable step up from the maximum conditions so far experienced in the Saxton core (14.5 - 15 Kw/ft) as less than two dozen rods of Core II would operate above 14.5 Kw/ft if the peak rod were to operate at 16 Kw/ft.

Because the Saxton reactor is an experimental plant, sustained periods of operation at the maximum rated power of 23.5 MWt have not been obtained in the past. The peak specific power in any fuel rod in Saxton is dependent on a great many factors; fuel enrichment, boron concentration, control rod position and reactor power level. Therefore, the peak specific power depends on the condition of the above parameters at the time the measurement is made.

With the reactor above 22 MWt, the maximum specific power level of the core is nominally 13-14 Kw/ft. This number is based on the same methods that would determine the 16 Kw/ft limit, that is, a 10% uncertainty in the measurements and an engineering hot channel factor of 1.045. The highest measured specific power has been 13.87 Kw/ft at a reactor power level of 22.9 MWt. When extrapolated to 23.5 MWt, a maximum of 14.56 Kw/ft is obtained from 12.16 Kw/ft at 19.63 MWt. With the uncertainties involved, it is not possible to say that with the reactor at 23.5 MWt that specific powers in excess of 14.5 Kw/ft have been experienced in the Saxton core. All of the peak values referred to above have occurred in the central 9 x 9 which contains experimental fuel that is licensed to operate up to 16 Kw/ft.

Successful operation of fuel at or above this level has been demonstrated by several Westinghouse experiments. Six capsules containing three fuel rod samples from the CVTR core were irradiated in the Westinghouse Test Reactor to a maximum power rating of 24 Kw/ft.[1] The capsule configuration was a 5-inch column of UO_2 pellets, .430 inches in diameter, 94 $\pm$ 1.5% of theoretical density clad with Zircaloy-2. The capsules were all successfully irradiated with no evidence of central melting.

Two additional capsules were irradiated in the Westinghouse Test Reactor.[2] One capsule contained three fuel rods with a 38-inch fuel length and was irradiated at peak fuel rod power levels of 17 to 19 Kw/ft to a maximum fuel burnup of 3,450 $\frac{MWD}{MTU}$. The other capsule contained four fuel rods with 6-inch fuel length. Average fuel rod power levels of > 18 Kw/ft were maintained during irradiation to 6,250 $\frac{MWD}{MTU}$. The rods contained UO_2 pellets .430 inches in diameter and 94 $\pm$ 1.5% dense. The capsules were clad in Zircaloy-2. The capsules were successfully irradiated and indicated that thermal reactors could be operated at these high rod powers safely and successfully.

UO_2 fuel capsules are being irradiated in the NASA - Plum Brook Reactor as part of the High Power, High-Burnup Irradiation Program.[3] Fuel pins containing 0.3 inch diameter pellets 96% dense with a 6-inch fuel column are clad with 304 stainless steel. The capsules are being irradiated at power ratings of 20 to 60 Kw/ft, to a maximum burnup of 80,000 $\frac{MWD}{MTU}$. Four capsules have been irradiated to 10,000 $\frac{MWD}{MTU}$ at a peak power rating of 39 Kw/ft. Three of these irradiations were completely successful; the fourth failed due to excessive fuel melting. Approximately seventy-five percent of the cross-sectional area of the pellets was molten. The failure occurred after long exposure at high rod power.

Three capsules were irradiated in the Plum Brook Reactor in a program designed to measure the thermal conductivity of UO_2 at the columnar grain growth threshold temperature.[3] The pins were 4-1/2 inches long and 1-1/4 inches in diameter. They were successfully irradiated at rod powers of 20-24 Kw/ft.

Two vibratory compacted pins and one pelleted fuel pin were successfully irradiated in the GETR at peak rod power of 21 Kw/ft.[4] The pins were 5.2 inches long and had an active fuel diameter of .56 inches. The pelleted rod was 88.3% dense while the vipac were 81.8% and 86.7%.

In addition, GE has run some very extensive, long irradiation high power level experiments in the GETR with fuel enriched to $\sim$ 20% in Pu.[5] Two pelletized rods with no central voids were operated at peak specific powers of $\sim$ 15.5 Kw/ft and $\sim$ 17.8 Kw/ft for burnup of 23,100 $\frac{MWD}{MTU}$ and 17,600 $\frac{MWD}{MTU}$ respectively. The experiments were very successful with no adverse effects due to these operating conditions.

Based on the experimental evidence available, the possible operation of some rods in the Pu region of Core II at 16 Kw/ft power levels will present no significant safety problems in the operation of Core II and is a very conservative extrapolation from the power levels already experienced in Saxton.

References

1. Duncan, R. N., "Rabbit Capsule Irradiation of UO_2," CVNA-142, 1962.

2. Duncan, R. N., "CVTR Fuel Capsule Irradiation," CVNA-153, 1962.

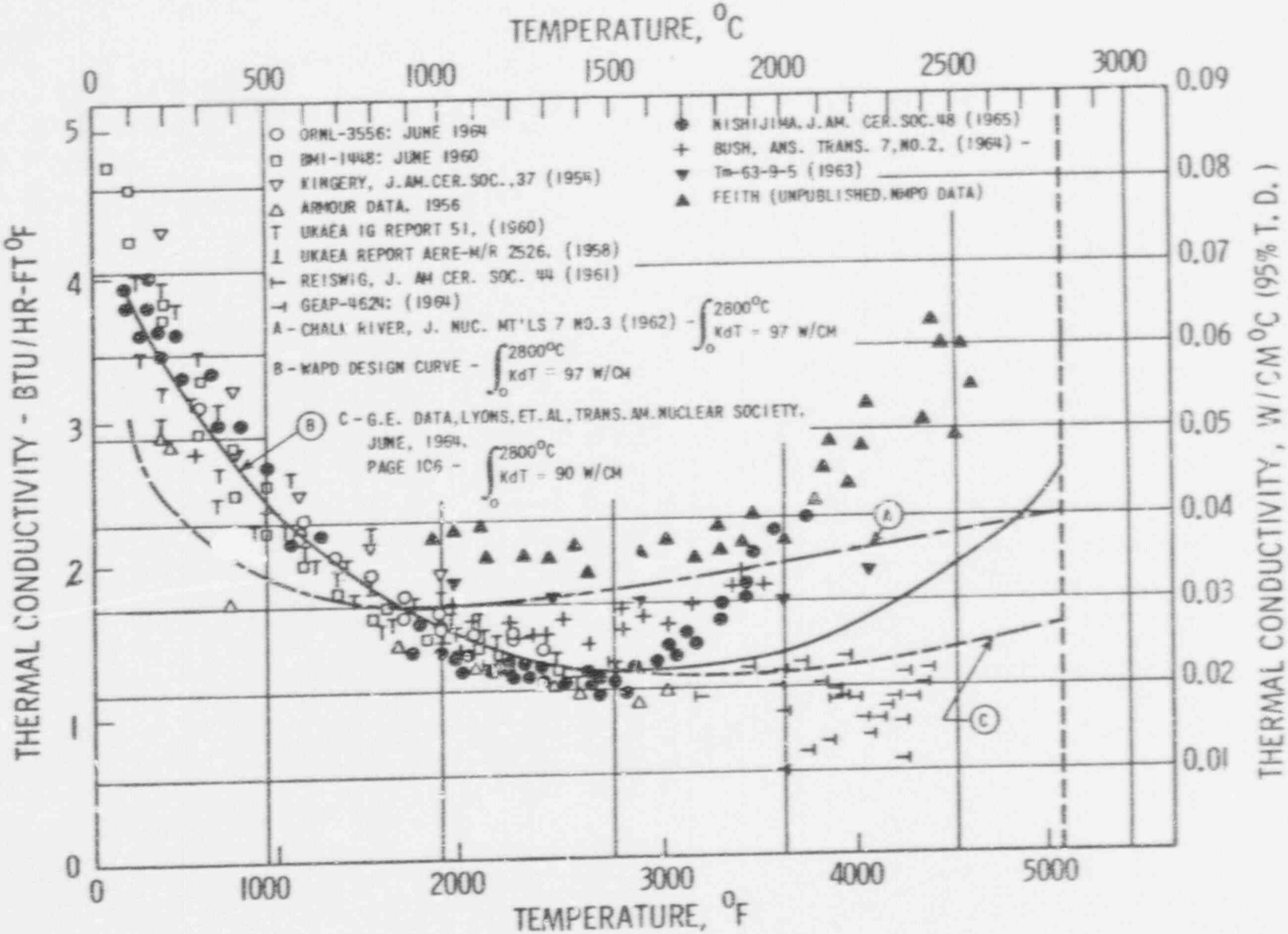
3. WCAP-2500, 2640, 2689, 2732.

4. Balfour, M. G. and Ferrari, H. M., "Irradiation of Vibratory Compacted UO_2 Fuel Elements," WCAP-2729.

5. Gerhart, J. M., "The Post-Irradiation Examination of a PuO_2-UO_2 Fast Reactor Fuel," GEAP-3833, 1961.

Question #3 - We understand that new information concerning the conductivity of uranium dioxide at high temperatures is available. Provide a curve of uranium dioxide conductivity as a function of temperature on which these new data points are included.

Answer: The attached figure is to replace Figure III-7 of the Core II Safeguards Report.



THERMAL CONDUCTIVITY OF URANIUM DIOXIDE

FIGURE III - 7

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Question #4 - The Saxton reactor is the first licensed nuclear power reactor in which a plutonium core loading is to be used. To enable us to evaluate a possible manner in which plutonium might be released to the environs, provide a discussion of those operating procedures which will assure that plutonium which may be in the containment building as contamination will not be transported to the remainder of the site or to the environs.

In addition, discuss why the limits of sensitivity of the various monitoring equipment and health physics procedures proposed are adequate to assure that 10 CFR 20 limits for plutonium will not be exceeded.

Answer: Because of the conservative assumptions and methods used in the plutonium fuel design and the rigorous testing and inspection performed on the fuel during its manufacture, the probability of fuel clad failure throughout the planned life of Core II is very small. In addition, the fuel rods and the fuel assemblies are monitored for alpha contamination prior to shipment to Saxton so that there is little likelihood that tramp plutonium will cause a contamination problem during fuel storage and loading.

In the event some plutonium contamination should be present inside the containment, there are only three methods available for transporting plutonium contamination from the containment building:

(a) Personnel

Saxton's present radiation protection procedures have proven adequate to prevent the spread of contamination from the

containment vessel. Access to the containment vessel is allowed only under the provisions stipulated by a radiation work permit which specifies, among other things, protective clothing to be worn. Step-off pads and storage for protective clothing are provided in the air lock. Monitoring of personnel for alpha contamination prior to leaving the vessel will be accomplished as required in the radiation work permit.

(b) Ventilation Exhaust

Since the containment vessel has no exhaust flow during reactor operation, the installed alpha monitoring system which will be added to the present containment air activity monitors will give a reliable history of containment vessel air activity. At a time when entry is desired, the reactor will be shut down and the containment vessel air activity will be known. Ventilation exhaust flow rate will be adjusted, if necessary, to insure that any release to the atmosphere is within the limits established by 10 CFR 20. It is expected that the containment vessel air activity attributable to plutonium will be below its MPC at all times and that it will not be necessary to regulate the containment vessel air release rate.

(c) Liquid Effluents

Liquid effluents from the containment vessel will be handled without any change to the present waste disposal or chemistry sampling system. The only procedural change will be an increased monitoring of areas for alpha contamination. Present procedures for monitoring effluents are adequate to assure that 10 CFR 20 limits for plutonium will not be exceeded.

After discussions between Saxton personnel and personnel at the Plutonium Recycle Test Reactor, we have concluded that the problems associated with radiation protection due to plutonium are no different from those which already exist, due to the presently installed uranium fuel. As quoted from U. S. Atomic Energy Commission Research and Development Report HW-83601, PROGRESS IN PLUTONIUM UTILIZATION by Hanford Laboratories:

> "Plutonium fuels have been stored and handled in the same manner as uranium fuel, and irradiated fuels have been routinely handled for special examinations and core changes without difficulty. No unusual procedural controls have been made necessary, nor has any specialized operator training been required specifically as a result of using plutonium fuels in the PRTR.
>
> "The PRTR experience has shown that the effects of plutonium fuel failures are no different than those for uranium fuels. Emissions have been virtually limited to fission gases with no evidence of particulate washout. Alpha contamination, usually of primary concern in fabricating plutonium fuels, is of little concern in reactor operation, as gamma contamination governs procedures for almost all maintenance work."

The activity concentration requirement of 10 CFR 20 for Pu-239, Pu-240 and Pu-241 for radiation workers exposed for 40 hours per week, is a maximum airborne concentration of 2.0×10^{-12} μc/cc. This activity level, defined as the radioactivity concentration guide for a 40 hour week (RCG/40), represents that concentration of plutonium in air to which a "standard man" may be exposed for 40 hours per week, 50 weeks per year for a total period of 50 years so that at the end of 50 years the total activity fixed in

the "standard man's" body will not exceed the recommended maximum permissible body burden (MPBB) of 0.04 μc of plutonium.

This MPBB as set by both the International Commission on Radiological Protection and the National Commission on Radiological Protection is defined as that amount of material which may be maintained indefinitely in the body of a "standard man" without producing any significant somatic or genetic effects throughout the life of the "standard man".

The sensitivities of the air particulate monitors, both the moving filter vapor container monitor and the fixed filter portable monitors, have been revised slightly from those given in the Core II Safeguards Report. The minimum sensitivity for these instruments for a 1-hour sample period and following a delay period (about 6 hours) to remove the Radon-Thoron background is given as 2.5×10^{-12} μc/cc. As stated before, containment access is not possible during power so that detection of this level of activity is more than adequate to assure that containment vessel purge prior to entry will not produce off-site plutonium levels above 10 CFR 20 levels. Containment vessel purge procedures can be altered, if required, if the containment vessel concentration is significantly above the limit of detection. Purge of the containment vessel prior to entry will assure adequate working conditions upon entry.

The portable air particulate monitors can be moved throughout various areas of the plant as required to sample for airborne activity. Alpha monitoring during such operations as main coolant sampling in the sample room or analysis work in the radiochemical laboratory is provided by these instruments. These instruments are capable of detecting near RPG/40 levels

with the 6-hour delay for Radon-Thoron decay. A more rapid readout of higher concentrations may also be obtained. Following a one-hour sampling time and the presence of a high Radon-Thoron background (600 cpm) the minimum sensitivity is about 2×10^{-10} μc/cc which is a factor of 100 above RPG/40. If this high plutonium concentration were detected, work in the area could be suspended and corrective action initiated. Workers exposed to these higher than RPG/40 concentrations could be restricted from working in possibly contaminated areas for a period of time to allow averaging of this exposure. For example, a one-hour exposure to 100 x RPG/40 concentration is equivalent to about 2-1/2 working weeks at RPG/40 so that return to work with RPG/40 concentrations would be permissible after 2-1/2 weeks of no exposure to plutonium.

Higher concentrations of plutonium can be detected in even shorter periods of time due to the fact that the count rate of the sample, due to Pu, increases linearly with exposure time and is proportional to the concentration. For a high Radon-Thoron background of 240 cpm a plutonium concentration of 1×10^{-9} μc/cc can be detected after a five minute sample time. Exposure to 1×10^{-9} μc/cc or 500 x RPB/40 for five minutes is almost equal to a 40-hour exposure to RPG/40, so that one week of non-exposure to plutonium would then allow return to work in RPG/40 levels.

The procedure of curtailing work following exposure to levels above RPG/40 is a standard practice and where combined with the instrument sensitivities described will assure that personnel exposures are well within the limits of 10 CFR 20.

Question #5 - In the accident analysis section of the report it is stated that each accident was analyzed using that combination of system parameters which would give the most serious consequences. Indicate the manner in which it can be assured that the most adverse combination of parameters has been selected, and provide the range of parameters considered for each accident analysis.

Answer: Two basic premises which underly accident and reactor transient analyses are to develop realistic yet conservative models and then to apply these models using realistic yet conservative parameters. Analog computers are normally used to simulate the reactor. The selection of the basic parameters depends on the transient being studied. The parameters are chosen on the basis of adding the most reactivity to the transient or providing the least help in limiting or preventing the transient.

As a specific example, the detailed reasoning for the choice of parameters of the control rod withdrawal at power accident are outlined below.

During this transient, heating of the fuel and the moderator will add negative reactivity to the systems and tend to depress the transient. For this reason, the moderator coefficient assumed was smaller than the expected value and would correspond to a boron concentration in excess of 2000 ppm. The Doppler coefficient chosen was less than expected values.

Overpower scram initiation is set to trip at 115% of nominal full power and is a redundant circuit to assure reliability. However, errors in fixing set points and in power measurements are assumed to delay scram initiation until a power level of 122% is reached.

Upon initiation of scram, an instrumentation delay of 0.5 sec. is assumed to delay rod motion. Actual instrumentation delay times are less than 0.3 seconds. A further delay in scram of 0.6 seconds is assumed for control rod motion in a region of small effectiveness and 0.9 seconds is assumed for completion of the rod insertion into the core. Actual measured control rod drop times for Saxton are on the order of 0.9 seconds or less so the actual scram completion time will be about 1.2 seconds or less compared to the 2.0 seconds assumed in the analysis.

Control rod scram worth upon insertion was assumed as 0.02 $\Delta k/k$. The nominal operating conditions of this accident, that is early in life with large hot channel factors and high boron concentrations (1500-2000 ppm), will result in about 0.15-0.18 $\Delta k/k$ reactivity in control rods out of the core. Even if the most reactive rod (0.05 $\Delta k/k$) were to stick, the reactivity insertion by control rods would be about 0.10 $\Delta k/k$. The only time that a reactivity insertion on the order of 0.02 $\Delta k/k$ would be possible would be very early in core life at very low boron concentrations (rodded control) which is a condition not compatible with the moderator coefficient chosen for the analysis.

A final conservative assumption is in the reactivity insertion rate of the control rods during withdrawal. The maximum insertion rate of the most reactive rod group (the two inner rods or the four outer rods) is 7.25×10^{-5} $\Delta k/k$/sec. and assumes the control rods to be in the most reactive region and moving at the maximum withdrawal speed. The value of 2.5×10^{-4} $\Delta k/k$/sec. which was assumed for this analysis is a much larger rate than could possibly be experienced by the reactor during this transient.

The same general reasoning has been applied to the other transients and accidents analyzed. The following tables present a comparison of the parameters assumed for the analyses and those which might be expected to exist in the reactor.

I. Rod Withdrawal, Cold Startup

	Value Used	Expected Value
1. Moderator Temperature Coefficient (at 70°F, 2000 ppm boron)	$+ 0.3 \times 10^{-4}$ Δk/k/°F	0.0 Δk/k/°F
2. Doppler Coefficient	$- 1.1 \times 10^{-5}$ Δk/k/°F	$- 2.0 \times 10^{-5}$ Δk/k/°F
3. Reactor Subcritical by	0.02 Δk/k	> .05 Δk
4. Overpower Scram Initiation	122%	115%
5. Control Rod Drop Time	1.5 sec.	< 0.9 sec.
6. Scram Reactivity Insertion by Rods	0.02 Δk/k	0.1 - 0.15 Δk/k
7. Reactivity Insertion Rate	2.5×10^{-4} Δk/k/sec.	$< 7.25 \times 10^{-5}$ Δk/k/se

II. Rod Withdrawal, Hot Startup

	Value Used	Expected Value
1. Moderator Temperature Coefficient (at 530°F, 2000 ppm boron)	$- 2.7 \times 10^{-4}$ Δk/k/°F	$- 3.0 \times 10^{-4}$ Δk/k/°F
2. Doppler Coefficient	$- 1.0 \times 10^{-5}$ Δk/k/°F	$- 1.3 \times 10^{-5}$ Δk/k/°F
3. Thru 7. - Same as for Case I		

III. Rod Withdrawal, At Power

	Value Used	Expected Value
1. Moderator Temperature Coefficient	$- 2.7 \times 10^{-4}$ Δk/k/°F	$- 3.0 \times 10^{-4}$ Δk/k/°F
2. Doppler Coefficient	$- 1.0 \times 10^{-5}$ Δk/k/°F	$- 1.1 \times 10^{-5}$ Δk/k/°F
3. Primary Coolant Pressure (For DNB Calculations) ΔH-DNB	2050 psi	2000 psi
Q-DNB	1950 psi	

III. Rod Withdrawal, At Power (Cont'd)

		Value Used	Expected Value-
4.	Instrument Delay Time	0.5 sec.	< 0.3 sec.
	Control Rod Drop Time	1.5 sec.	< 0.9 sec.
5.	Reactor Power Level, % of Nominal	103%	95-100%
6.	Overpower Scram In[illegible]	122%	115%
7.	Scram Reactivity In[illegible] by Rods	0.02 $\Delta k/k$	0.10-0.15 $\Delta k/k$
8.	Maximum Specific Pow[illegible]	16.5 Kw/ft	14-15 Kw/ft

IV. Steam Break

1.	Moderator Temperature Coefficient (Worst Case, End of Life - 0 ppm Boron Concentration)	-4.1×10^{-4} $\Delta k/k/^{\circ}F$	-4.0×10^{-4} $\Delta k/k/^{\circ}F$
2.	Safety Injection Functions	No	Yes

V. Loss of Flow Accident

1.	Moderator Temperature Coefficient	-2.7×10^{-4} $\Delta k/k/^{\circ}F$	-3.0×10^{-4} $\Delta k/k/^{\circ}F$
2.	Control Rod Drop Time	1.5 sec.	< 0.9 sec.
3.	Reactor Power Level - % of Nominal	103%	95-100%
4.	Scram Reactivity Insertion of Rods	0.02 $\Delta k/k$	0.10-0.15 $\Delta k/k$
5.	Maximum Fuel Power Density	16.5 Kw/ft	14-15 Kw/ft
6.	Primary Coolant Pressure (For DNB Calculations) $\begin{cases} \Delta H\text{-DNB} \\ Q\text{-DNB} \end{cases}$	2050 psi 1950 psi	2000 psi

Question #6 - In the report it is stated that the results of the chemical shim experiment program have demonstrated that a boron release accident as originally postulated is not credible and, accordingly, the requirements of an unexplained reactivity limit are no longer required. Provide a description of the results of the chemical shim work at Saxton so that we may evaluate the safety considerations of deleting this requirement.

Answer: To answer this question, copies of WCAP-2599, "The Saxton Chemical Shim Experiment," are submitted herewith.

Question #7 - Provide an estimate of the amount of plutonium that might be released to the containment in the event of the "maximum hypothetical accident" to enable a more definitive evaluation of the consequences of this accident. In addition, provide an evaluation of the amount of plutonium that might subsequently reach the environs.

Answer: A conservative evaluation of the amount of plutonium oxide in the containment vessel following the maximum hypothetical accident has been completed. The maximum amount of PuO_2 that could be in the containment vessel would be less than 50 mg and maximum amount available for leakage in the form of an aerosol would be less than 35 mg. These amounts would result in a maximum two hour inhalation exposure at the site boundary of less than 10^{-8} of the permissible body burden for plutonium.

Evaluation of the maximum hypothetical accident for the Saxton reactor partial plutonium Core II considered a condition in which the emergency systems to provide core cooling did not function following a loss-of-coolant accident. For such a situation, decay heat generated in the core will result in extensive melting of the clad and internal supports and will eventually cause the core to collapse into the bottom of the reactor vessel. This situation will expose a large amount of fuel surface to the atmosphere in the reactor vessel and the high temperatures involved will cause volatilization of the fuel.

The amount of fuel which can be volatilized under these circumstances will be severely limited because of the geometry of the system, the presence of an air atmosphere and the fact that the fuel may be partially wetted by the molten clad or even partly submerged in a pool of molten cladding and structures.

As shown in Figure 7-1, experimental evidence[1,2] indicates that the vapor pressures of plutonium dioxide and uranium dioxide follow the same curve as a function of the reciprocal of the absolute temperature as measured in a vacuum. Also shown on this figure are the experimental data[3] for the vapor pressure of PuO_2 in an air atmosphere. As would be expected, the presence of an air atmosphere reduces the vapor pressure below that measured in a vacuum. For this calculation, it will be assumed that PuO_2 and UO_2 have the same vapor pressure - temperature relationship in an air atmosphere.

An empirical relationship has been developed which correlates the weight loss rate, vapor pressure, absolute temperature and molecular weight for a system vaporizing a substance in an insulated crucible with a small opening. The relationship is as follows:[4]

$$P_{(atm)} = 6.267 \times 10^{-9} \quad \mu/Ka \sqrt{\frac{T}{M}} \qquad (1)$$

P = partial pressure of the effusing species, atm

μ = weight loss rate - mg/hr

K = Klausing factor $[K = 1/(1 + 0.5\ L/R)]$

a = effective orifice area - cm^2 (730 cm^2)

T = absolute temperature - °K

M = molecular weight

L = orifice length (assumed as 1 ft.)

R = orifice radius (1/2 ft.)

The Klausing factor is applied because the actual orifice has some finite physical dimensions while the correlation was developed for an ideal orifice. The molecular weight of the fuel will be taken as an average of 271. Using these constants, Eq. (1) becomes:

$$\mu = \frac{P}{\sqrt{T}} \times 9.59 \times 10^{11} \text{ mg/hr} \qquad (2)$$

If an average temperature of ~ 2400°K is assumed for the core material which is slowly heating throughout the meltdown, the corresponding pressure is ~ 10^{-6} atm. The weight loss rate is then:

$$\mu = \frac{10^{-6}}{2400} \times 9.59 \times 10^{11} \text{ mg/hr}$$

$$\mu = 1.96 \times 10^{4} \text{ mg/hr}$$

$$\mu = 19.6 \text{ g/hr}$$

The PuO_2 in the core is 6.6 w/o of the central nine assemblies. As there are 21 assemblies in the core, the average PuO_2 w/o is $6.6 \times \frac{9}{21} = 2.5$ w/o. If it is assumed that the volatilized material has the same weight fraction of PuO_2, then the

$$\mu_{PuO_2} = 0.49 \text{ gm/hr.}$$

The value of $\mu_{PuO_2} = 0.49$ gm/hr would be the limiting value if the entire reactor vessel were at the temperature assumed for the hot fuel as was the case in the experiments of Reference (4). Most of the reactor vessel will be at temperatures considerably lower (500-600°F) than the 4000°F used for the average of the fuel mixture. Because of this situation, a great deal of the vaporized fuel material will not leave the reactor vessel but will plate-out on the relatively cold internal surfaces of the vessel.

The surface area of the inside of the reactor vessel which might be available for plate-out is estimated at about 2.5×10^5 cm^2. The cross-sectional area of the main coolant pipe is about 730 cm^2 so that the ratio is about 3×10^{-3}. Therefore, a conservative estimate of the rate at which the vaporized plutonium oxide leaves the break would be 1% (three times the area ratio) of the rate calculated by Equation (2). The rate at which PuO_2 leaves the vessel is therefore 4.9 mg/hr. In the unlikely event that a condition of no core cooling were to occur, it is not expected that it would exist for more than a few hours so that the total amount of PuO_2 release to the containment vessel would be less than 50 mg.

Because of the large amount of relatively cold surface available in the containment vessel for plate-out of the volatilized material, it is not expected that there will be any significant airborne concentration of PuO_2 which might cause an inhalation hazard. As an upper limit on the evaluation, it will be assumed that all of the PuO_2 leaving the reactor vessel is of the proper particle size to remain in the containment atmosphere as an aerosol.

Studies[5] on the reduction rate of the mass concentration of aerosols indicates that a half life of 4-5 hours is typical. Assuming a half life of 5 hours, an equilibrium state for the amount of PuO_2 in aerosol form is soon reached. The equilibrium amount is calculated as follows:

$$\frac{dN}{dt} = R - \lambda N \tag{3}$$

N = amount of PuO_2 in the aerosol, mg

R = release rate - mg /hr

λ = decay constant - hr^{-1}

Solution of equation (3) yields the familiar result

$$N_{(t)} = \frac{R}{\lambda}\left[1 - e^{-\lambda t}\right] + N_{(o)}e^{-\lambda t} \tag{4}$$

At equilibrium with $N_{(o)} = 0$

$$N_{eq} = \frac{R}{\lambda} \tag{5}$$

As shown before R = 4.9 mg/hr

$$\lambda = \frac{.693}{5}$$

$$N_{eq} = \frac{4.9 \times 5}{0.693} = 35.4 \text{ mg}$$

An equilibrium amount of 35.4 mg PuO_2 gives a total weight of Pu of 31.2 mg. Assuming that the Pu aerosol has the same isotopic concentrations as were present in the fuel, we have 2.7 mg of Pu-240 and 28.5 mg Pu-239. These weights give activities of 0.6 x 10^{-3} curies of Pu-240 and 1.77 x 10^{-3} curies of Pu-239 or a total of 2.37 x 10^{-3} curies of Pu.

The original off site inhalation hazards for the Saxton maximum hypothetical accident have resulted in a Technical Specification containment leak rate limit of 0.4% of the contained volume per day. This leak rate is based on a design pressure of 30 psig existing throughout the accident. The design pressure was

based on the total energy release of the reactor coolant at saturated water conditions and 2000 psi. The actual energy content of the reactor coolant is considerably less than that assumed previously. Also, Figure 506.1 in the Final Hazards Report for Saxton indicates that the containment pressure will drop very rapidly from the initial peak.

Using the generalized Gaussian dispersions equation[6] for a ground level point source and assuming Pasquill type "F" conditions with a wind speed of 1 meter per second, a dispersion factor $\frac{\chi \bar{u}}{Q} = 6 \times 10^{-3}\ m^{-2}$ is obtained at the exclusion radius of 300 meters. Additional credit[6] can be taken because of dispersion and dilution in the wake of the containment building so that $\frac{\chi \bar{u}}{Q}$ becomes:

$$\frac{\chi \bar{u}}{Q} = \frac{1}{\pi \sigma_y \sigma_z + cA} \qquad (\pi \sigma_y \sigma_z = 167\ m^2)$$

cA = building dilution factor

A = building cross section = 250 m^2

c = factor ranging from 0.5 to 2 depending on the building, assumed as 0.5

Therefore:

$$\frac{\chi \bar{u}}{Q} = 3.42 \times 10^{-3}\ m^{-2}$$

Fall-out of particles as the plume travels will also provide additional reduction of the plume concentration. This reduction factor can be estimated for this case using the method proposed by Chamberlain.[7]

The deposition reduction factor (DRF) is given by

$$(\text{DRF}) = \exp\left(-\frac{2}{\pi}\frac{v_g}{\bar{u}}\int_0^x \frac{1}{\sigma_z} e^{-\frac{1}{2}\left(\frac{h}{\sigma_z}\right)^2} dx\right)$$

For this case the release height, h = 0

$$(\text{DRF}) = \exp\left(-\frac{2}{\pi}\frac{v_g}{\bar{u}}\int_0^x \frac{1}{\sigma_z} dp\right)$$

For Pasquill "F" conditions $\int_0^{300} \frac{1}{\sigma_z} dx$ is about 300. Data from Stewart(8) indicates that the deposition velocity, v_g, for plutonium oxide with particle sizes to be expected in the aerosol size range is in the range of 3-5 cm/sec. If a value of v_g = 4 cm/sec is chosen then:

$$\text{DRF} = \exp\left(-\frac{2}{\pi}\frac{4}{100} \times 300\right)$$

$$\text{DRF} = \exp(-7.64)$$

$$\text{DRF} = 4.92 \times 10^{-4}$$

The plume concentration of Pu at the site boundary is then given by:

$$\chi_{300} = Q \times 3.42 \times 10^{-3} \times 4.92 \times 10^{-4}$$

$$Q = 2.37 \times 10^{-3} \times \frac{4 \times 10^{-3}}{24 \times 3600} = 1.1 \times 10^{-10} \text{ c/sec}$$

$$Q = 1.1 \times 10^{-4} \ \mu\text{c/sec}$$

$$\chi_{300} = 1.85 \times 10^{-10} \ \mu\text{c/m}^3$$

If an active adult breathing rate of 1.25 m^3/hr and an uptake retention factor of .25[9] are assumed, the two hour uptake of Pu is:

$$D_{Pu} = 1.85 \times 10^{-10} \times 2 \times 1.25 \times .25 = 1.16 \times 10^{-10}\ \mu c$$

The maximum permissible body burden of Pu is 0.04 μc[10] so the accident uptake is 2.9 x 10^{-9} below the permissible body burden. Because of the large deposition fraction within the exclusion radius, there will be no significant plutonium released beyond the site boundary.

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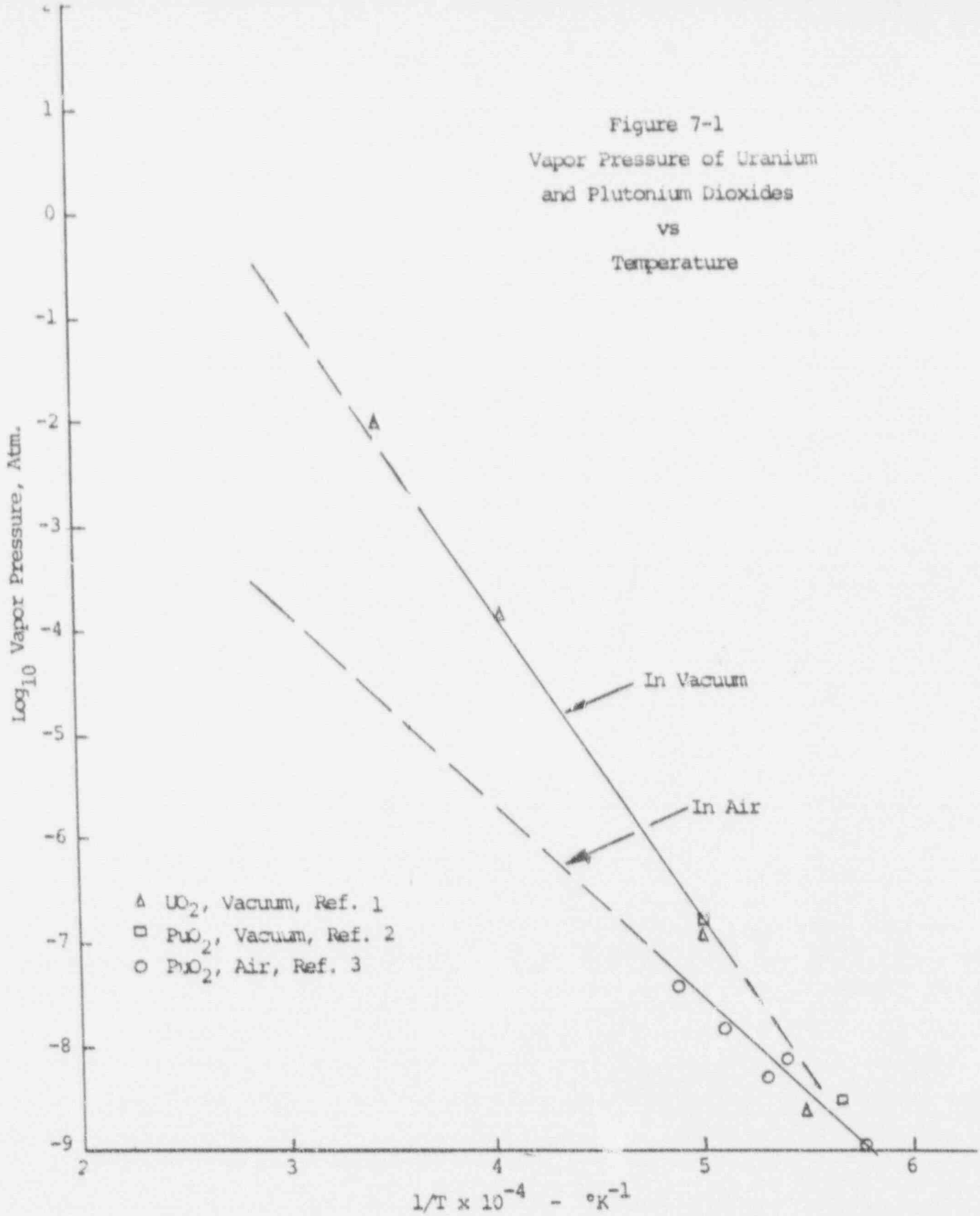

Figure 7-1
Vapor Pressure of Uranium
and Plutonium Dioxides
vs
Temperature
In Vacuum
In Air
UO_2, Vacuum, Ref. 1
PuO_2, Vacuum, Ref. 2
PuO_2, Air, Ref. 3
Log_{10} Vapor Pressure, Atm.
$1/T \times 10^{-4}$ - $°K^{-1}$
1
0
-1
-2
-3
-4
-5
-6
-7
-8
-9
2
3
4
5
6

Question #8 - Since plutonium requires somewhat more stringent consideration of the reactivity requirements for fuel storage than uranium, provide an evaluation of the adequacy of the Saxton fuel storage facilities for plutonium fuel.

Answer: Evaluation of the adequacy of the Saxton fuel storage facilities for the plutonium enriched fuel were carried out using PDQ-3 calculations to determine the subcritical multiplication factors of the UO_2 and PuO_2-UO_2 fuel assemblies when installed in the fuel storage racks.

The physical dimensions of the fuel storage racks consist of a 3.2-inch surface-to-surface fuel element separation in each row and a 12-inch separation between rows. Ambient water temperature conditions with 0 ppm of boron were assumed for the calculation although the fuel storage water is actually borated. The results of the calculations are shown below:

Fuel	Calculated k_{eff}
UO_2	0.838
PuO_2-UO_2	0.898

The calculated k_{eff} for the PuO_2-UO_2 fuel includes a correction to account for the discrepancy between the experimental results of the WREC criticals and the predicted analytical results. From the data in the above table, it is concluded that there will be no criticality problems or hazards in storing either type of fuel assembly at Saxton.