ASSESSMENT OF SHUTDOWN SYSTEM TRIP PARAMETER EFFECTIVENESS FOR CANDU REACTORS FOLLOWING IN-CORE LOSS OF COOLANT ACCIDENTS

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1.0 INTRODUCTION

Assessment of shutdown system trip parameter effectiveness is performed as part of the detailed analysis of postulated in-core loss of coolant accidents (LOCAs). In-core LOCAs can arise, for example, from simultaneous failure of a pressure tube/calandria tube, severe flow blockage of a fuel channel, or as a result of a feeder break leading to channel flow stagnation. For most operating states, an in-core LOCA results in a reactor core and heat transport system response that is nearly identical to that of an out-of-core LOCA with the same break discharge rate. In particular, reactor power is well-controlled up to the time of reactor trip and trip coverage on each shutdown system is provided by parameters such as heat transport low pressure, heat transport low flow, and/or pressurizer low level (where applicable).

For plant operating states in which the moderator contains a large amount of soluble neutron poison (*e.g.*, startups), an in-core LOCA can result in a significant insertion of positive reactivity to the system as unpoisoned heat transport (HT) coolant displaces poisoned moderator fluid. The increase in moderator temperature also contributes to an increase in reactivity. This positive reactivity effect may be compensated in part if the isotopic purity of the coolant is less than that of the moderator. The moderator poison concentration is highest in the earliest stages of startup following a long shutdown because of the need to compensate for the decay of short-lived fission product poisons. Additional poison is also required when the core is in the pre-equilibrium state, to offset any additional excess reactivity associated with fuel at low irradiation. The reactivity insertion may result in an increase in reactor power prior to reactor trip, depending upon the response of the reactor regulating system. The overall analysis comprises detailed simulations of moderator response, reactor core response (including reactor regulating system [RRS] behaviour), system thermal hydraulics and fuel behaviour for a wide range of in-core LOCA scenarios. This paper will specifically address the assessment of trip parameter effectiveness for in-core LOCAs. Emphasis is placed on the approach taken to synthesize and extend the results of the detailed analysis over the entire spectrum of plant operating states. Detailed assessment of reactor core response following in-core LOCAs is discussed in Reference 1. Typical results of assessments performed for Ontario Hydro's CANDU reactors are provided together with specific examples of operating limits and design changes identified to ensure adequate trip coverage.

2.0 TRIP PARAMETER EFFECTIVENESS ASSESSMENT

2.1 Trip Effectiveness Criteria

Trip parameter effectiveness assessment is performed to provide assurance that timely shutdown system intervention will occur over the entire range of possible reactor conditions. For small break LOCA events, trip parameters are considered to be effective if they preclude fuel sheath failure and fuel channel failure.

Fuel sheath integrity is maintained if:

(a) the local strain anywhere on the sheath remains less than 15 percent,
(b) the maximum fuel temperature remains below the melting point (*i.e.*, 2840·C), and
(c) athermal strain remains less than 0.4 percent.

Precluding fuel melting is also a sufficient (although not essential) condition for maintaining fuel channel integrity.

Extensive analysis of post-dryout fuel behaviour during small LOCA transients indicates that sheath failure may not be precluded if sheath temperatures exceed 600·C for significant periods of time. A trip parameter can therefore be considered to be effective in a given small LOCA scenario provided that it initiates before sheath temperatures reach 600·C and fuel temperatures reach 2840·C. Analysis indicates that the sheath temperature criterion is generally more limiting than the fuel temperature criterion.

2.2 Overall System Response Characteristics for In-Core LOCAs

The net rate of positive reactivity insertion following an in-core LOCA, and therefore the overall system response, will depend on a number of factors including:

(i) the moderator poison concentration,
(ii) the extent to which the HT coolant is less isotopically pure than the moderator,
(iii) the break discharge rate, and
(iv) the response of the RRS.

Power increases prior to reactor trip may mask or delay the process trips and accelerate the time to onset of dryout and subsequent fuel overheating following the break. On the other hand, power increases will enhance the effectiveness of the neutronic trips. RRS-induced reactivity device movement due to changes in reactivity following the break may introduce spatial power distortions which can also accelerate the onset of dryout, even if no bulk power increase is experienced.

Figure 1 provides a conceptual illustration of the various regimes of system behaviour associated with an in-core LOCA. The vertical scale corresponds to the initial operating power level whereas the horizontal scale represents net positive reactivity insertion rate. High reactivity rates correspond to high moderator poison concentration and low moderator-to-coolant isotopic purity differences. Low rates correspond to low moderator poison concentrations and/or high isotopic purity differences (*i.e.*, with coolant purity significantly below that of the moderator).

With reference to Figure 1, three distinct regimes of system behaviour can be identified for in-core LOCA events:

(1) Very Low (~0) Net Reactivity Insertion Rate: These conditions exist most of the time when operating at steady state under equilibrium fuelling conditions. Following the break, the RRS is able to maintain reactor power essentially constant prior to trip and overall system response is similar to that associated with an out-of-core break. Trip coverage for all break sizes is provided by the HT low pressure, HT low flow, pressurizer low level (where applicable) or high moderator level (where installed) trips.

(2) High Power/High Reactivity Insertion Rate: At high initial power levels, above approximately 40 to 50 percent full power (FP), the bulk power excursion prior to trip (typically at a maximum rate of 1 percent to 2 percent FP/s) may mask or delay the pressurizer low level (where applicable) and HT low pressure trips. This phenomenon can potentially occur for the most rapid power increases due to the tendency of the HT coolant to swell as its enthalpy is increased, thereby counteracting the tendency of the HT system to depressurize due to the HT coolant inventory loss following the break. System behaviour under these conditions becomes typical of that associated with slow loss of reactivity control events. Effective trip coverage is provided only by trips on Neutron Overpower (NOP) and (where installed) high moderator level. Regulatory requirements normally mandate that two independent and diverse trip parameters be effective for all operating conditions. Therefore, in cases where the high moderator level trip is not installed, the NOP trip alone is effective and there is a potential gap in trip parameter coverage.

(3) Low Power/Low-to-Intermediate Reactivity Insertion Rate: At low initial power levels, below about 40 to 50 percent FP, the relative increase in power prior to trip would be essentially the same as that associated with breaks occurring from high power. However, due to the lower initial power level, the absolute rates of increase (in percentFP/s) are lower. In addition, HT system temperatures (and therefore the coolant saturation pressure) are initially lower at lower power levels. Thus, immediately following the break, the HT system will initially depressurize rapidly to the relatively low saturation pressure, after which the rate of depressurization is determined by the rate of mass inventory loss and the rate of power increase associated with the break. At low initial power levels, the combined effect of lower coolant saturation pressures and low absolute rates of power increase help ensure that the HT low pressure trip and (where applicable) the pressurizer low level trips remain effective. Trip coverage for this case is therefore similar to that associated with out-of-core LOCAs, despite the fact that reactor power increases prior to trip initiation.

Other regions in Figure 1 exhibit behaviour that lies between that associated with the three regimes identified above. This transitional behaviour requires detailed assessment to precisely delineate the range of conditions over which a given trip parameter can be considered to be effective.

2.3 Trip Parameter Assessment Methodology

The analysis of in-core LOCAs requires a complex series of calculations to develop trip coverage maps which summarize the range of power level, break size, moderator poison concentration and moderator-coolant isotopic purity difference over which each of the trip parameters is effective.

The key parameters of interest in the assessment of shutdown system trip parameter effectiveness over the range of reactor operating conditions are the process and neutronic trip times, the time at which onset of sheath dryout is predicted to occur, and the sheath temperature following the onset of dryout. As indicated in Figure 2, a number of simulation codes are employed in the analysis of trip parameter effectiveness for in-core LOCAs:

SMOKIN- A three-dimensional modal neutron kinetics code which permits detailed simulation of the transient behaviour of the spatial neutron flux distribution. The code permits calculation of individual channel powers, total reactor power, Neutron Overpower detector readings, and contains a detailed model of the reactor regulating system. The SMOKIN model and the assumptions employed in the analysis are described in more detail in Reference 1.

SOPHT- A semi-implicit finite difference code designed for simulation of the transient thermal hydraulic behaviour of the heat transport system (HTS). It contains a comprehensive flow network model, together with built-in controller and component models, including a point kinetics model and a simplified model of the reactor regulating system (Reference 2). For most in-core LOCA analysis, the code employs total reactor power transients generated with the more detailed SMOKIN code. The code permits calculation of the response of HTS pressure, temperature and voiding rate, together with core flow and (where applicable) pressurizer level. The code is also used to evaluate the rate of mass and energy discharge from the break into the moderator.

MINI-SOPHT- A simplified version of the SOPHT code, used with transient boundary conditions at the reactor headers obtained from the main SOPHT simulation and with individual channel power transients from SMOKIN. The header-to-header channel models of selected channels are used to evaluate the earliest time at which the onset of dryout could potentially occur. The shutdown system flow instrumented channels are also modelled in detail to permit the evaluation of HT low flow trip times. Each MINI-SOPHT channel model contains detailed representations of individual fuel channels, end-fittings and inlet and outlet feeders.

SOMASS- A code used to simulate the response of the moderator system following in-core breaks (Reference 3). Moderator temperature transients are calculated by performing a lumped parameter energy balance on the system. The moderator temperature response is used as input to the calculation of core reactivity balance following the break. The moderator level transient is evaluated by accounting for swell and for any addition to the initial moderator inventory due to the break. This permits the evaluation of trip time on the high moderator level trip, where applicable.

ELOCA-MK4S- A high temperature transient fuel behaviour code which is used to address the mechanisms and potential for sheath failure under accident conditions (Reference 4). Failure mechanisms associated with the different metallurgical zones of the sheath (*i.e.*, as-received, alpha-annealed and prior-beta) are considered. Post-dryout cooling conditions are supplied by a MINI-SOPHT simulation and channel power transients are obtained from SMOKIN. Initial fuel conditions are evaluated for potentially limiting combinations of element power and burnup by the ELESIM-II(MOD 10) code (Reference 5). The code permits an evaluation of the time following event initiation beyond which sheath failure cannot be precluded.

Significant assumptions made in the analysis of trip effectiveness for in-core LOCAs include:

(a) Initial channel thermal powers for fuel cooling assessments are based on the maximum instantaneous power likely to be experienced in a given channel under nominal steady state core conditions. These powers are derived from channel ripple data (fuelling history) or based on other constraints such as the licence limit maximum channel power or channel outlet temperature alarm limits, as applicable.

(b) No credit is taken in the analysis for any power reduction (or other mitigating effect) due to protective action from the reactor regulating system. However, reactivity device movement arising from the normal control response to high positive power error, such as moderator level reduction (Pickering A only) or insertion of mechanical control absorbers (all reactors other than Pickering A) is considered if the resulting flux distortions degrade trip effectiveness. Moreover, impairment of regulating system response due to device damage following the in-core break is considered if it leads to further degradation in trip coverage due to impairment of reactor power control or due to increased severity of the flux distortion.

(c) Effective trip setpoints, which include the effects of measurement loop uncertainties and simulation error allowances, are used in the analysis of trip parameter effectiveness. Trip timing is based on the time at which the effective setpoint is reached on the third-out-of-three SDS logic channels, to conservatively allow for the possibility that one channel in the two-out-of-three voting logic arrangement is unavailable.

(d) The HT low pressure trip and the pressurizer low level trip (where applicable), employs a dual level setpoint, with a lower setpoint applicable at low power. At Pickering NGS A and B, the power signal for setpoint switching purposes is derived from the out-of-core ion chambers. In-core detector signals are employed at Bruce NGS A and B and Darlington NGS. For accidents occurring below the setpoint switching power, automatic restoration of the setpoint applicable at high power is credited, provided the reactor power has risen above the setpoint switching power, and the power signal can be demonstrated to be unaffected by the initiating event.

(e) In the case of the HT low flow trip, each of the instrumented channels is assumed to be operating at their minimum permissable power. A low initial channel power delays trip initiation, and conservatively accounts for variations in initial channel power caused by fuel burnup and other possible deviations from the nominal time-averaged flux shape. Typically, a value ranging from 80 percent to 90 percent of the time averaged channel power is assumed.

2.4 Trip Coverage Assessment-Results

Results for Pickering A are discussed in detail to illustrate key features of the trip assessment methodology and results. In the case of Pickering NGS A at equilibrium fuelling conditions, the moderator poison load in the early stages of reactor startup following a long shutdown can reach 42 mk. The analysis considers the case in which moderator level control functions normally following the break, and the case in which moderator level control is not credited and the moderator level is assumed constant at its initial value. Moderator level control supplements the normal reactivity control function of the liquid zone controllers. These assumptions span the range of expected moderator level response following an in-core break from the viewpoint of core behaviour. (For other reactors, it should be noted that bulk reactivity control is supplemented by the insertion of mechanical control absorbers, and the analysis considers potential impairments of their function arising from the break). Trip coverage is evaluated as a function of initial reactor power level, and the moderator poison concentration, or alternatively, the difference in isotopic purity between the coolant and the moderator.

In the analysis for Pickering A, the difference in isotopic purity between the moderator and the coolant is no lower than 0.0 wt percent, which represents an operating limit. Normally, the isotopic purity of the coolant would be less than that of the moderator and the isotopic purity difference would be greater than 0.0 wt percent. Under these conditions, the associated degradation in moderator purity arising from the discharge of coolant into the moderator leads to a negative reactivity effect which offsets (in part) the positive reactivity associated with dilution of the heavily poisoned moderator. At Pickering A, an isotopic purity difference (moderator purity minus coolant purity) of 1 wt percent has the same effect on core reactivity balance following the break as a reduction of 24 mk in moderator poison load.

Analysis results are summarized in Table 1. Detailed results are provided for the maximum break size of 230 kg/s and an intermediate break size of 100 kg/s. Analysis is performed at two initial power levels, namely 103 percent FP and 40 percent FP. For the purposes of parametric analysis, system response is evaluated with progressively larger break-induced positive reactivity insertions. With the maximum moderator poison load of 42 mk, reactivity insertions associated with an isotopic purity difference of 0.0 wt percent, 1.0 wt percent and 2.0 wt percent were evaluated. Higher isotopic purity differences result in lower reactivity insertion rates. In Table 1, the 1.0 wt percent case is equivalent to assuming an isotopic purity difference of 0.0 wt percent, but with a poison concentration of only 18 mk. The 2.0 wt percent case is equivalent to assuming a poison concentration of 0 mk at an isotopic purity difference of 0.25 wt percent. Therefore, the analysis spans a range of isotopic purity difference, or equivalently, covers the entire range of moderator poison concentration.

Trip coverage for Pickering NGS A provided by the process and neutronic trip parameters is summarized in Figures 3a and 3b for the 230 kg/s break, assuming respectively that moderator level functions normally and that moderator level control is not credited. Trip coverage is shown as a function of initial power level and moderator poison concentration, assuming that isotopic purity difference is at the 0 wt percent operating limit. Coverage expressed in terms of moderator-to-coolant isotopic purity difference at the 42 mk maximum poison concentration is shown on the topmost scale of these figures. Figure 4 indicates the dependence of trip coverage on break size, for the case in which moderator poison load is 42 mk at an isotopic purity difference of 0 wt percent.

The trip parameters which provide protection for in-core LOCA events at Pickering NGS A are Neutron Overpower (NOP), HT low pressure (HTLP) and HT low flow (HTLF).

In the case of the NOP trip, Figure 5a shows the power transients measured by the third ion chamber signal to register a trip. Figure 5b provides the power transient in the maximum powered channel together with the predicted time to onset of dryout. For the 230 kg/s break, a moderator poison concentration of 42 mk yields the fastest power excursion. The associated power transient is similar to that which would arise from a net positive reactivity insertion rate of 0.005 mk/s. Rates of power increase are too low to initiate a trip on the High Log Rate parameter. For lower poison concentrations, the RRS is able to suppress the initial power excursion for a limited period of time. For the case with the most rapid increase in power, the power in the maximum power channel at the time that the detectors reach the effective trip setpoint is 106 percent of its initial value, whereas the dryout power for this fast transient is approximately 120 percent. As the initial reactor power level is reduced, the dryout power level is reduced due to the greater depressurization. At sufficiently low power levels, the NOP parameter is no longer effective, as shown in Figure 6a. This degradation in NOP coverage at low powers is evident in Figures 3a and 3b.

Figure 5c indicates the response of the reactor outlet header (ROH) pressure following a 230 kg/s break as a function of time. The effect of moderator poison concentration and initial power level is demonstrated in this figure. For a given moderator poison concentration, the ROH pressure reached during the transient decreases as the initial reactor power level decreases. This is a consequence of the depressurization rate (which is much more rapid when the ROH pressure is above the saturation pressure), and a lower saturation pressure at lower initial power levels. In general, due to the more rapid initial depressurization, the coverage provided by the HTLP trip improves at lower initial power levels, even in the presence of a significant reactor power excursion. Figure 6b shows the minimum ROH pressure reached (or, the ROH pressure at the time sheath temperature reaches 600·C) following a 230 kg/s break as a function of initial power, for various moderator poison concentrations. These curves are used directly to establish the range of initial power and moderator poison load over which the HTLP trip provides coverage. At 40 percent FP, the HTLP trip setpoints are switched to lower values; however, the analysis accounts for restoration of the setpoint applicable at power levels greater than 40 percent FP. This is indicated by the dashed setpoint line in Figure 6b.

Figure 5d indicates the response of the flow in the limiting flow-instrumented channel as a function of time, for in-core LOCA events occurring with a wide range of moderator poison concentration. For high values of moderator poison concentration, the power excursion accelerates the onset of two-phase conditions in the channel and the associated reduction in flow. At lower poison concentrations, the rate of increase in power is lower and the trip setpoint is reached later in the transient. The effectiveness of the HTLF trip for a given transient depends on the time beyond which fuel sheath integrity can no longer be assured. Figure 6c provides, as a function of initial reactor power, the time at which the low flow trip is registered on three-out-of-three safety channels and the time at which sheath temperature reaches the adopted criterion for sheath integrity of 600·C. Trip times are nearly invariant with reactor power whereas the time at which sheath temperature exceeds 600·C decreases at higher initial power levels due to the lower margins to dryout when operating at high power.The intersection of the timing lines for the low flow trip and 600·C sheath temperature establishes the maximum power level at which the trip is effective. As indicated in Figures 3a and 3b, the HTLF trip becomes effective over a narrower range of power levels at high moderator poison concentrations.

In summary, for Pickering NGS A, as indicated in Figures 3a and 3b, only the neutron overpower trip is effective for in-core LOCA events starting from high initial power levels when the moderator poison load is high. The analysis demonstrates that at least two independent trip parameters are effective under all other operating conditions.

This analysis was performed for all of Ontario Hydro's CANDU reactors. Results similar to those presented for Pickering NGS A were obtained, *i.e.*, for in-core LOCAs occurring with the moderator heavily poisoned and with isotopic purity of the coolant nearly equal to that of the modeartor, only the NOP trip is effective if the reactor operates at or near full power. Figure 7 shows the trip coverage map derived for shutdown system no.1 at Bruce NGS B. The most significant difference with respect to Pickering NGS A (which uses feed and bleed type pressure control rather than a pressurizer) is the added coverage provided at low power by the pressurizer low level trip and the moderator high temperature trip.

3.0 MEASURES ADOPTED TO ENSURE ADEQUATE TRIP COVERAGE

Regulatory requirements applicable to CANDU reactors such as those embodied in R-10 and R-8 (References 6 and 7) mandate that for each accident, two diverse and independent trip parameters must be demonstrated to be effective on each shutdown system over the entire range of permitted plant operating states. In order to ensure that this requirement would be met at each of Ontario Hydro's CANDU reactors, a number of operating restrictions and design modifications were implemented following completion of the in-core LOCA analysis.

Reactor Power Limits as a Function of Moderator Poison Concentration

As indicated in Figures 3a, 3b and 7, coverage by at least two trip parameters can be assured for in-core LOCAs by avoiding high power operation at high poison concentrations. This is achieved in practice by placing limits on reactor power as a function of moderator poison concentration (refer to Figure 8). The most restrictive limit on power will exist at the earliest stages of the startup process following a long shutdown when moderator poison is highest. As the startup process proceeds and saturating fission product poisons build-in, moderator poison can be removed, thereby permitting a further reactor power increase. Eventually, sufficient poison is removed to permit operation at full power. It should be noted that the limits are somewhat less restrictive if the moderator-to-coolant isotopic purity differece is higher at the time of startup. This is because, as explained in Section 2.4, an increase in isotopic purity difference has the same effect as a reduction in moderator poison load. Limits of this type have been implemented at Pickering NGS A and B and Bruce NGS A and B. These limits are accompanied by limits on minimum isotopic purity difference.

Trip Parameter Modifications

In lieu of the operating restrictions described above, implementation of a high moderator level trip can provide effective coverage for in-core LOCA events, even when operating at high reactor power levels with a high moderator poison concentration. This approach was adopted for Darlington NGS A, where shutdown system no. 1 already had a trip on high moderator level, and implementation of the same trip parameter on shutdown system no.2 required only a minor change to the trip computer software due to the existence of a moderator low level trip. As a result, full two parameter trip coverage was provided for all operating states and no special operating limits were required during the startup process.

CONCLUSION

Analysis of trip parameter effectiveness for in-core LOCA events identified potential gaps in trip parameter coverage when operating at high reactor power with a heavily poisoned moderator. A thorough assessment was performed to delineate the range of operating conditions over which each trip parameter is effective. This permitted the formulation and successful implementation of procedural and/or design changes to ensure compliance with the applicable regulatory requirements.

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Table 1
Summary of Shutdown System Reactor Process Trip and NOP Trip Initiation Times and Dryout Times Following PT/CT Failure (Equilibrium Fuel Conditions)

Case Description		Moderator Level Control	NOP (1) (s)	HTLP (s)	HTLF (2) (s)	Dryout Time (s)	Sheath Temperature Reaches 600°C (s)
230 kg/s Initial Break							
103% FP	0% I.D. (3)	WML (4)	6	NR (6)	106	36	55
	1% I.D.	WML	31	115	104	60	80
	2% I.D.	WML	114	83	119	113	143
40% FP	0% I.D.	WML	>150	1	80	155(7)	>155
	1% I.D.	WML	>150	1	88	230(7)	>230
	2% I.D.	WML	>150	1	109	>150	>150
103% FP	0% I.D.	NML (5)	6	NR	>80	39	48
	1% I.D.	NML	31	NR	>120	62	72
	2% I.D.	NML	108	86	123	110	129
40% FP	0% I.D.	NML	115	1	>80	101(7)	>101
	1% I.D.	NML	>120	1	105	146(7)	>146
	2% I.D.	NML	>150	1	108	>150	>150
100 kg/s Initial Break							
103% FP	0% I.D.	WML	26	NR	312	74	104
	1% I.D.	WML	72	288	278	124	169
	2% I.D.	WML	350	203	318	339	>350
40% FP	0% I.D.	WML	295	4	339	280	304
	1% I.D.	WML	>480	4	243	431	>450
	2% I.D.	WML	>480	4	320	>450	>450
103% FP	0% I.D.	NML	24	NR	>194	69	88
	1% I.D.	NML	71	NR	242	127	154
	2% I.D.	NML	341	203	318	337	379
40% FP	0% I.D.	NML	174	4	NR	192	213
	1% I.D.	NML	284	4	NR	291	313
	2% I.D.	NML	>480	4	332	>450	>450

(1) NOP trip time based on channel power peaking factor of 1.28, consistent with the evaluation of dryout time.

(2) The HTLF trip is based on an instrumented channel power of 80% of the time-averaged channel power.

(3) I.D. - Moderator-to-Coolant Isotopic Purity Difference : weight% (i.e., Moderator Purity minus Coolant Purity). Moderator poison load of 42 mk assumed.

(4) WML - with moderator level control.

(5) NML - no moderator level control.

(6) NR - Not reached up to the end of the simulation period, well after the predicted time of onset of dryout.

(7) Based on extrapolation past end of the simulation period.

Figure 1

Regimes of System Behaviour Associated With an In-Core LOCA

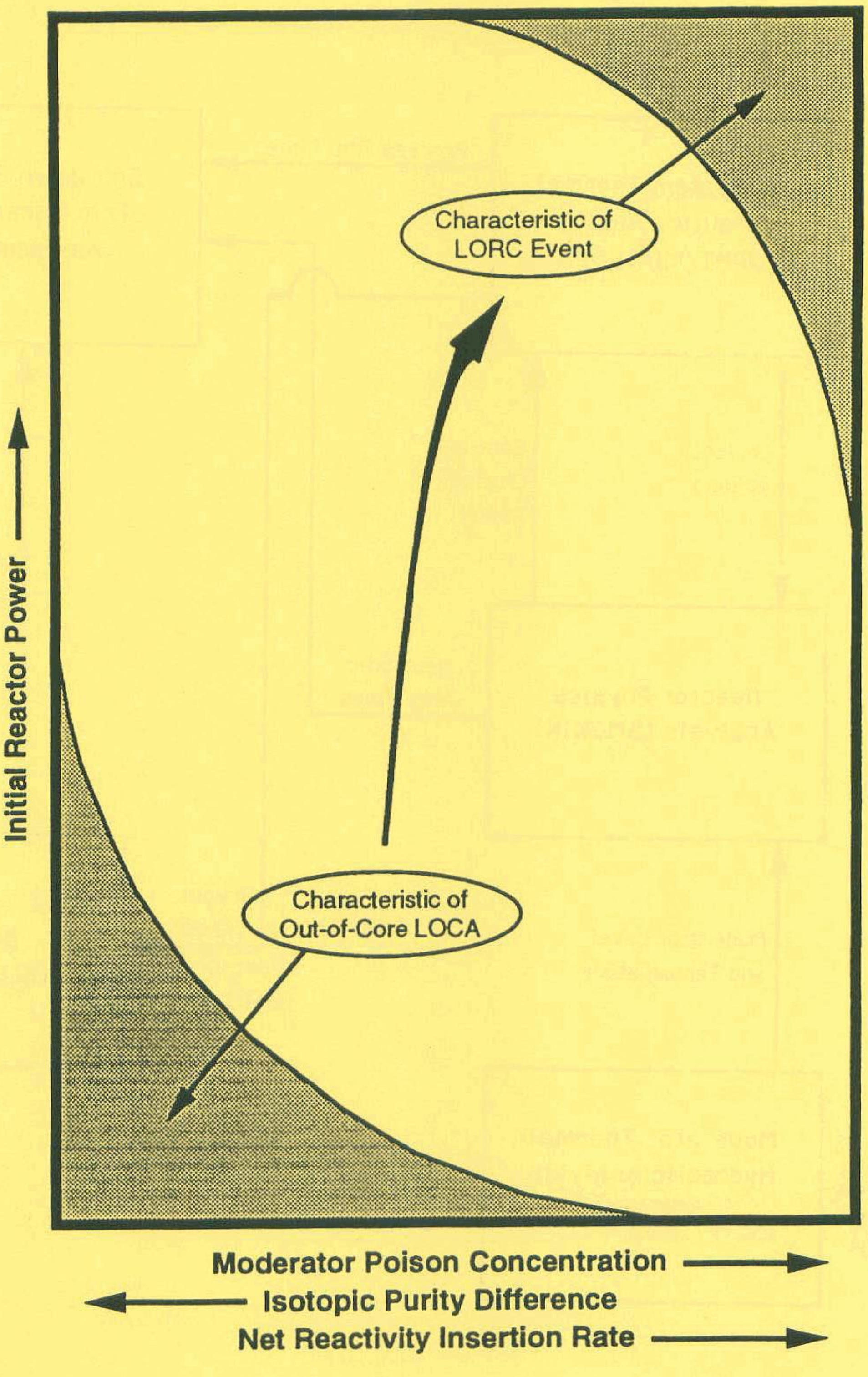
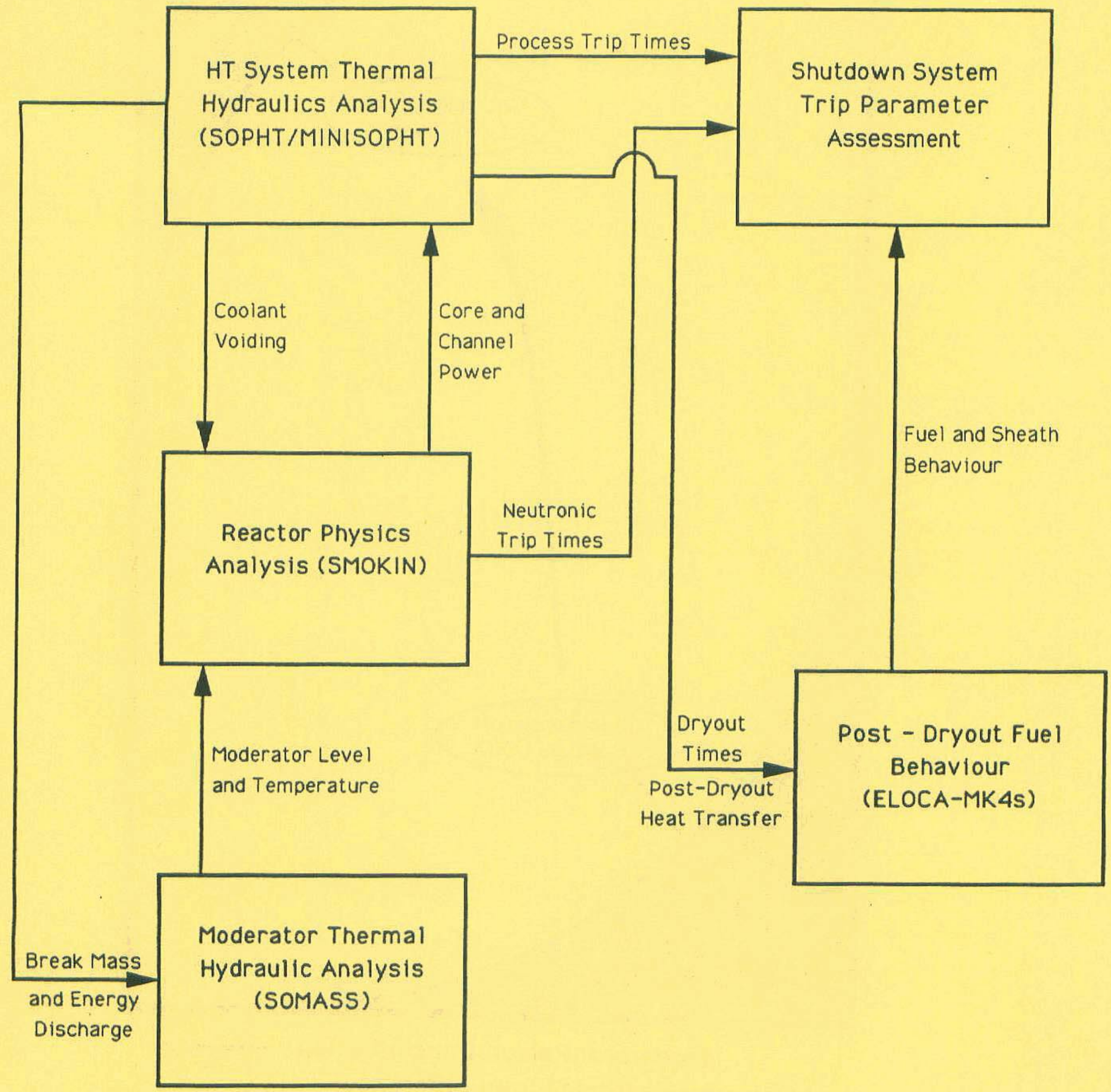


Figure 2

In-Core LOCA Analysis
Process For Trip Assessment



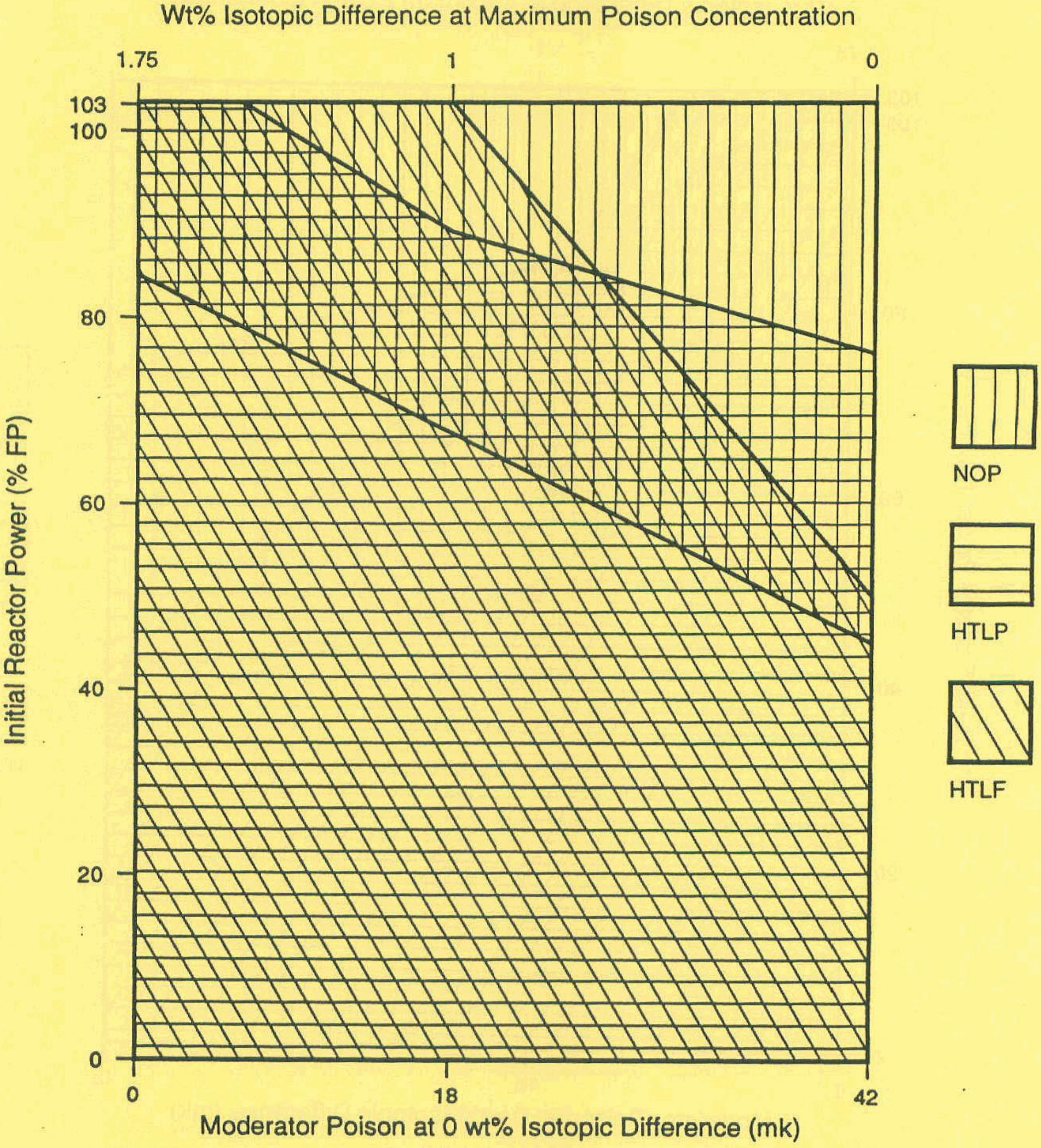


Figure 3(a)

SDS Trip Parameter Effectiveness During 230 kg/s In-Core LOCA Under Full HT Flow Conditions With Moderator Level Control Functioning Normally

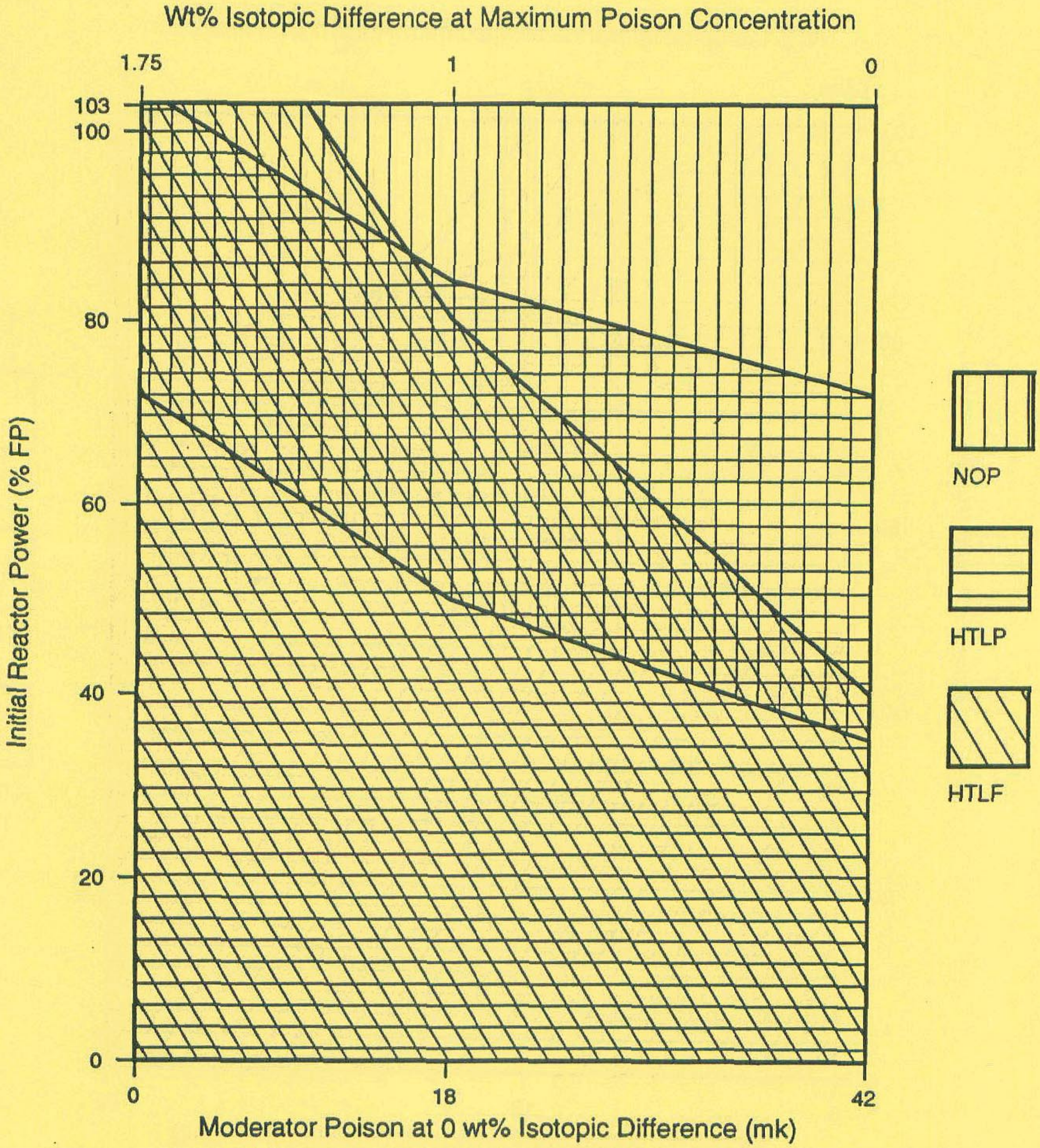


Figure 3(b)

SDS Trip Parameter Effectiveness During 230 kg/s In-Core LOCA Under Full HT Flow Conditions With No Moderator Level Control

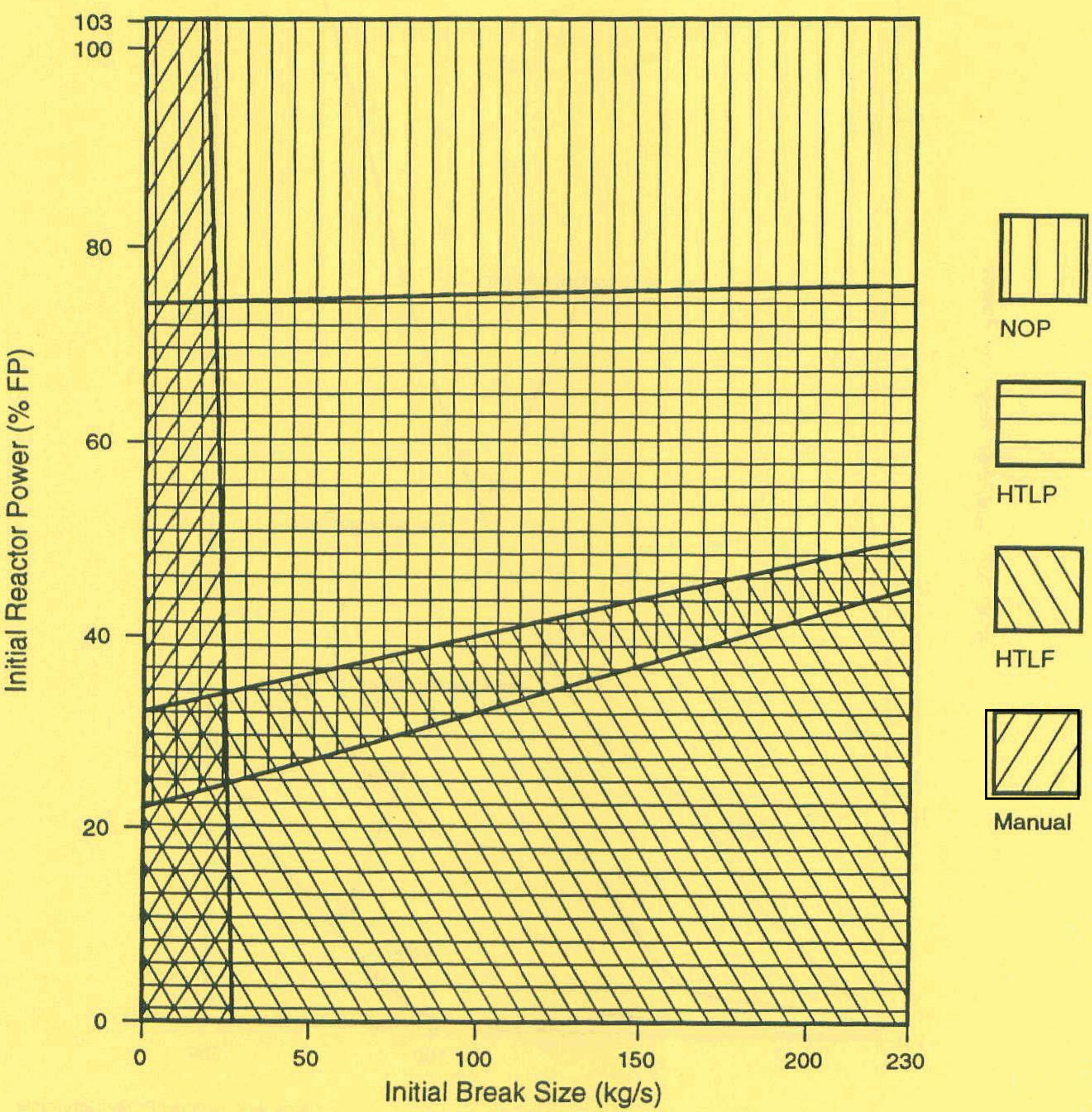


Figure 4

SDS Trip Parameter Effectiveness for In-Core Break
Under Full HT Flow Conditions at 0 wt% Isotopic Difference
at Maximum Startup Poison Concentration (With Moderator Level Control)

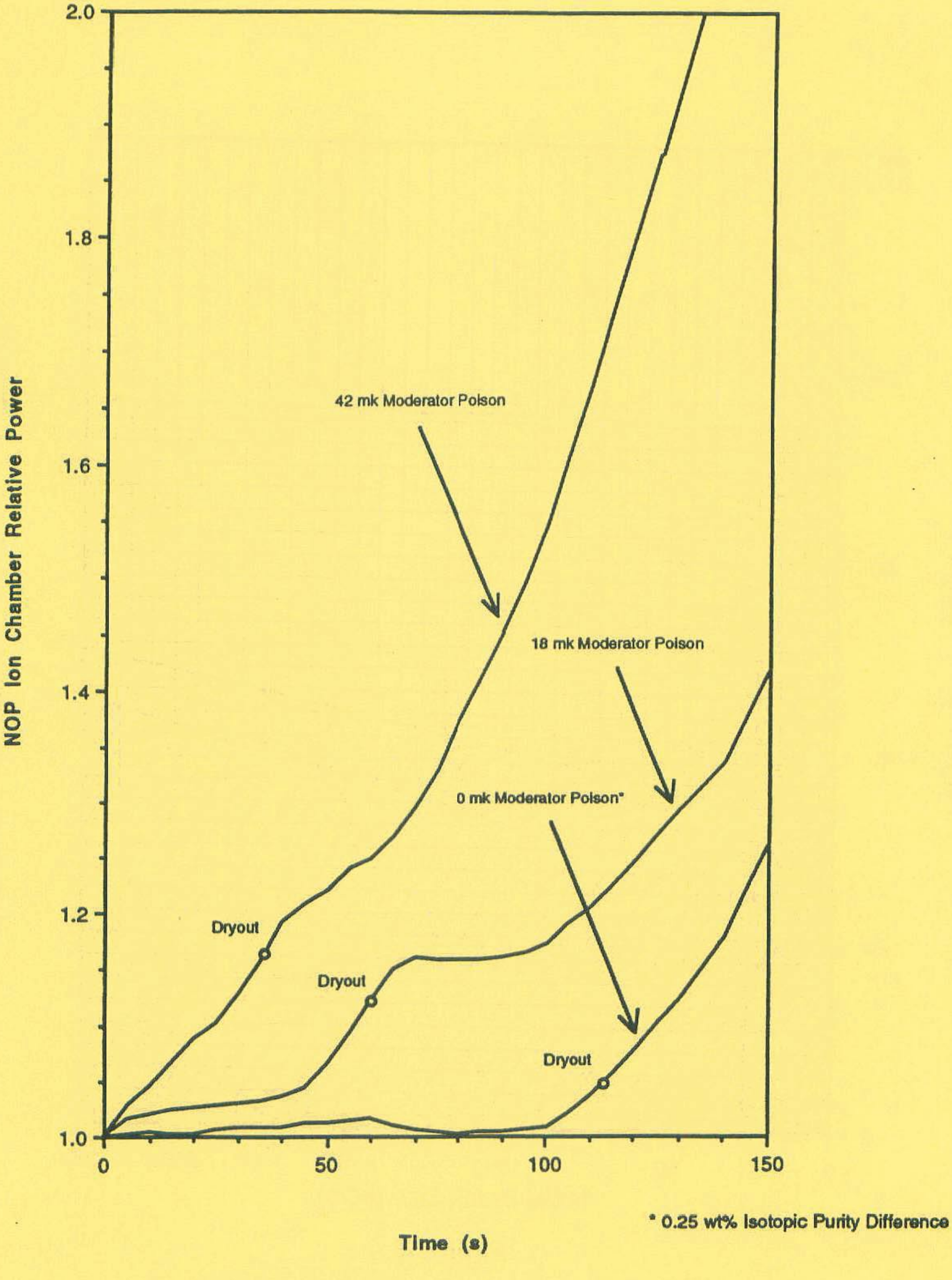


Figure 5(a)

SDS NOP Relative Signal Transient During 230 kg/s In-Core LOCA With 0 wt% Isotopic Purity Difference from 103% FP Under Full HT Flow Conditions (With Moderator Level Control - No Reactor Trip Credited)

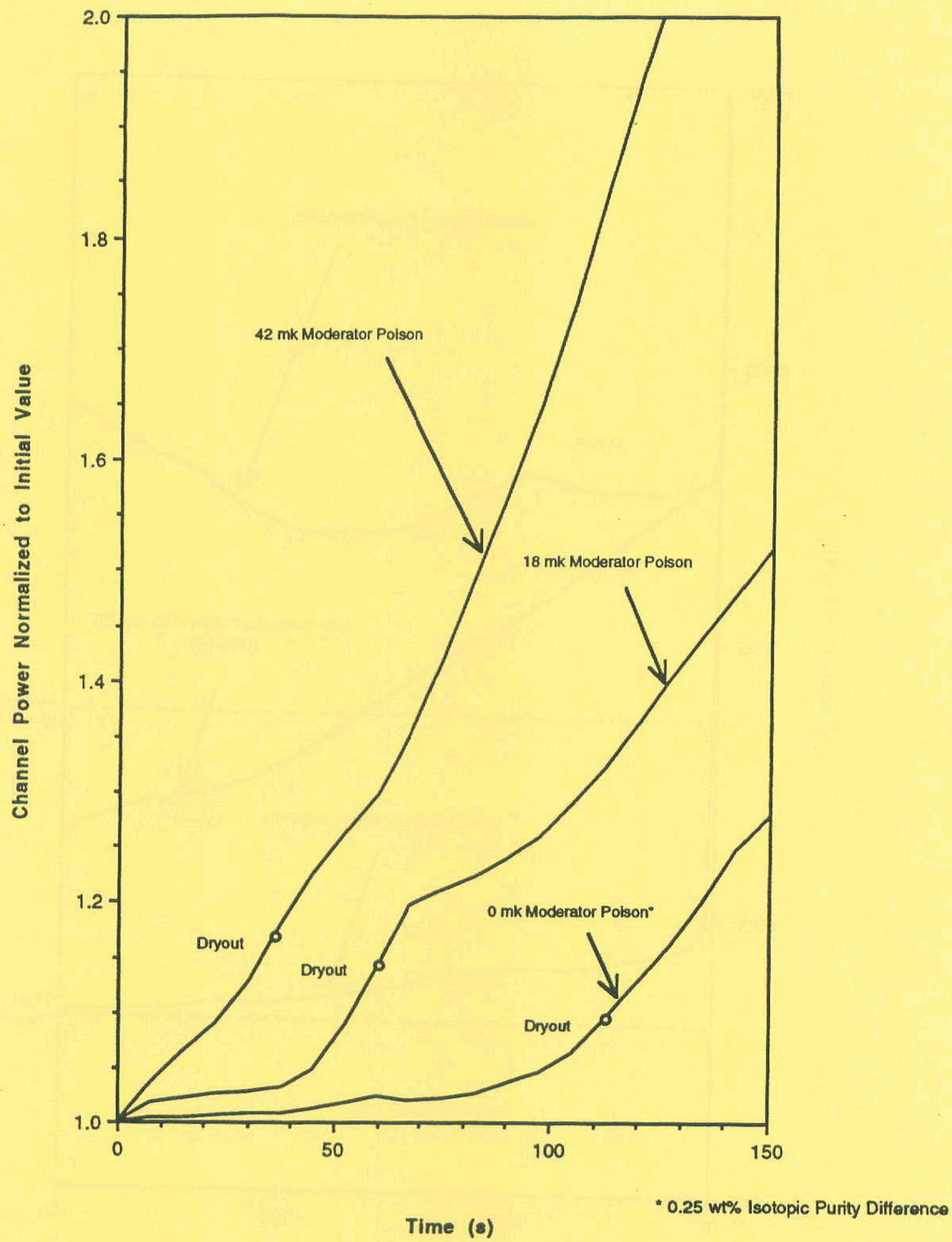


Figure 5(b)

Maximum Channel Power Transients During 230 kg/s In-Core LOCA With 0 wt% Isotopic Purity Difference from 103% FP Under Full HT Flow Conditions (With Moderator Level Control - No Reactor Trip Credited)

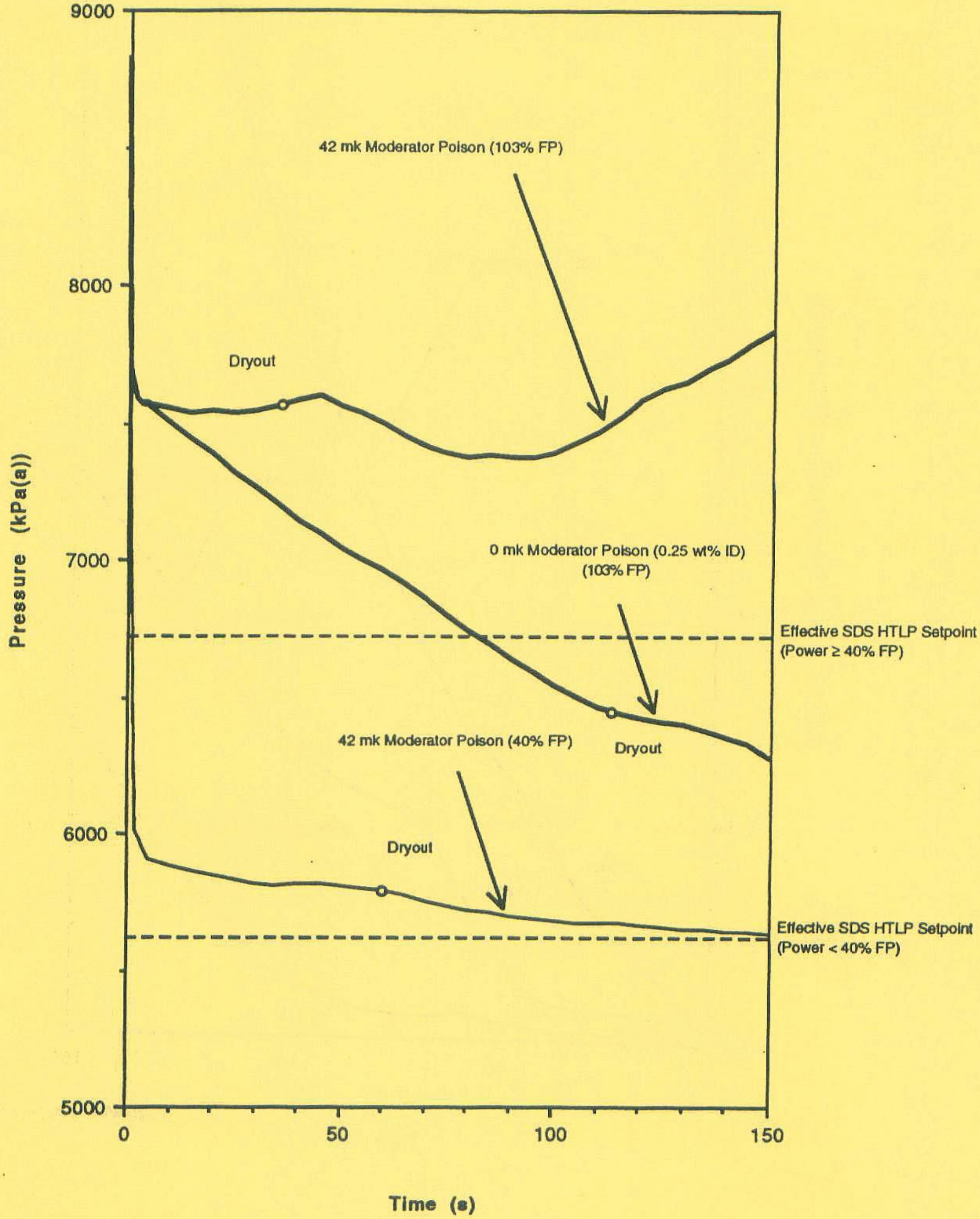


Figure 5(c)

ROH Pressure Transients During 230 kg/s In-Core LOCA With 0 wt% Isotopic Purity Difference Under Full HT Flow Conditions (With Moderator Level Control - No Reactor Trip Credited)

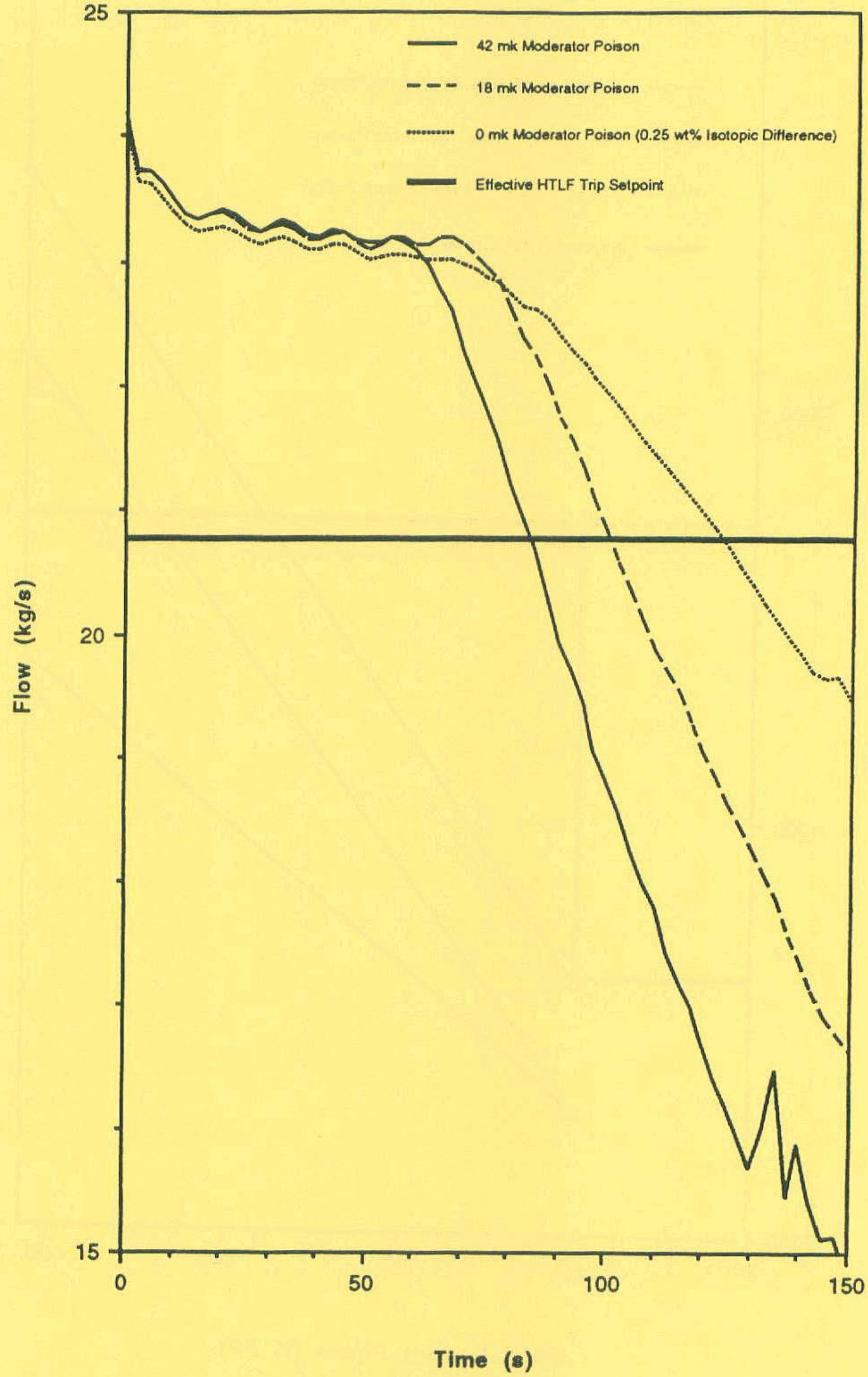


Figure 5(d)

Flow-Instrumented Channel Flow Transients as a Function of Initial Moderator Poison Concentration During a 230 kg/s In-Core LOCA With Moderator Level Control Functioning Normally (103% FP Initial Reactor Power / Equilibrium Fuel)

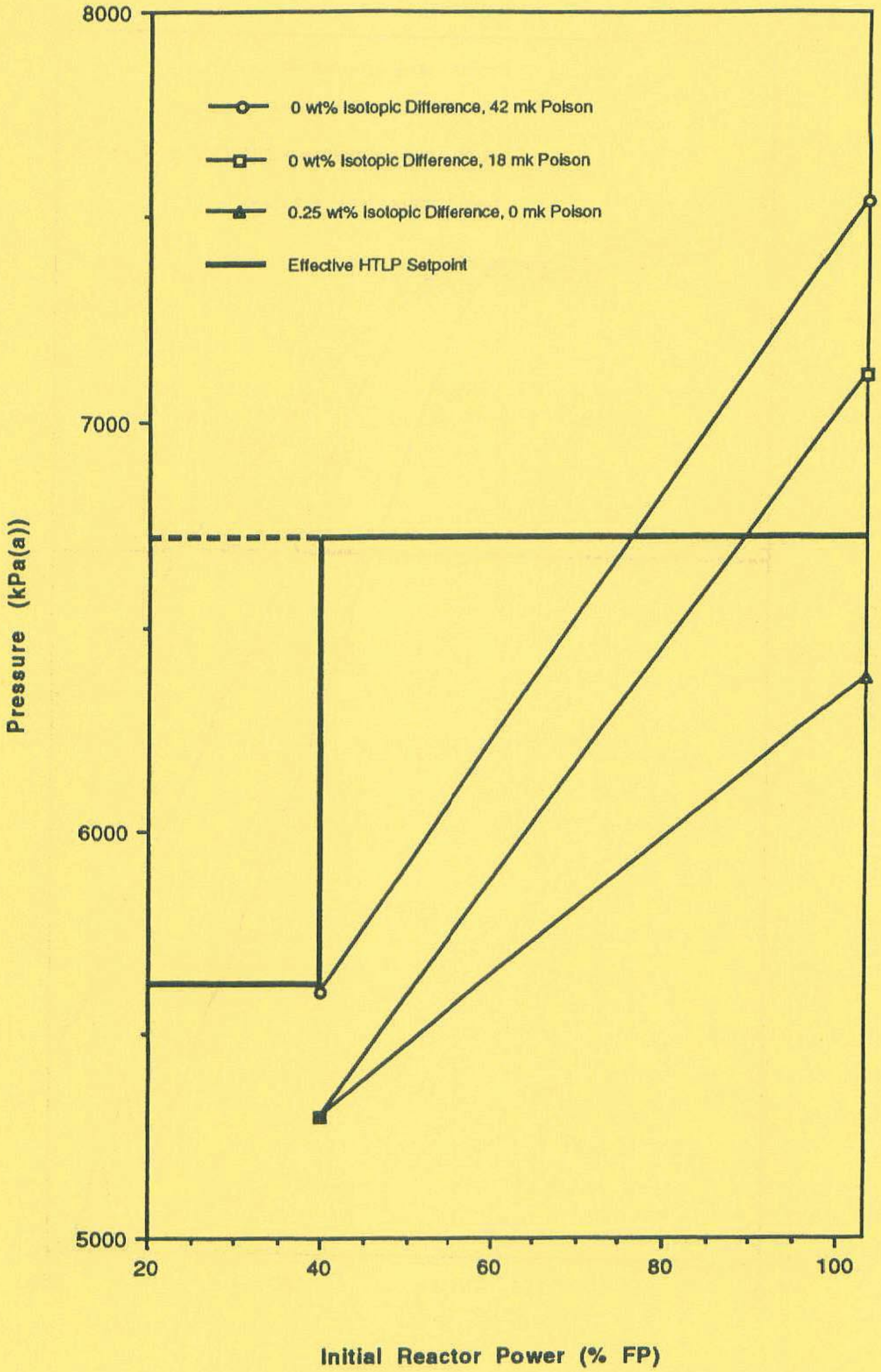


Figure 6(a)

Minimum ROH Pressure Attained or Pressure at Time Fuel Sheath Temperature Reaches 600°C During 230 kg/s In-Core LOCA (Equilibrium Fuel / With Moderator Level Control)

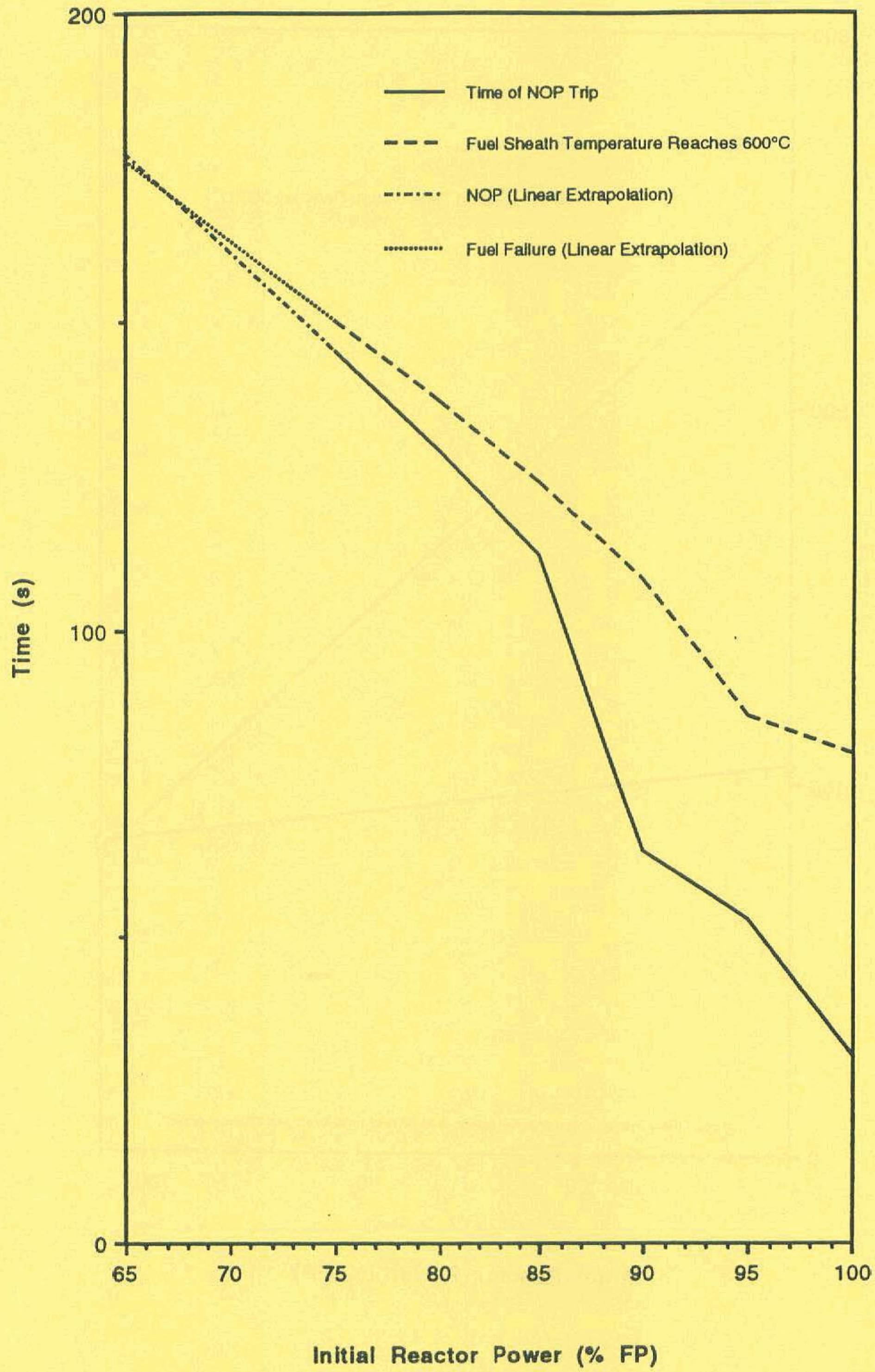


Figure 6(b)

Time of NOP Trip and Time that the Fuel Sheath Reaches 600°C as a Function of Initial Reactor Power Following a 230 kg/s In-Core Break (0 wt% Isotopic Difference 18 mk Moderator Poison / Equilibrium Fuel)

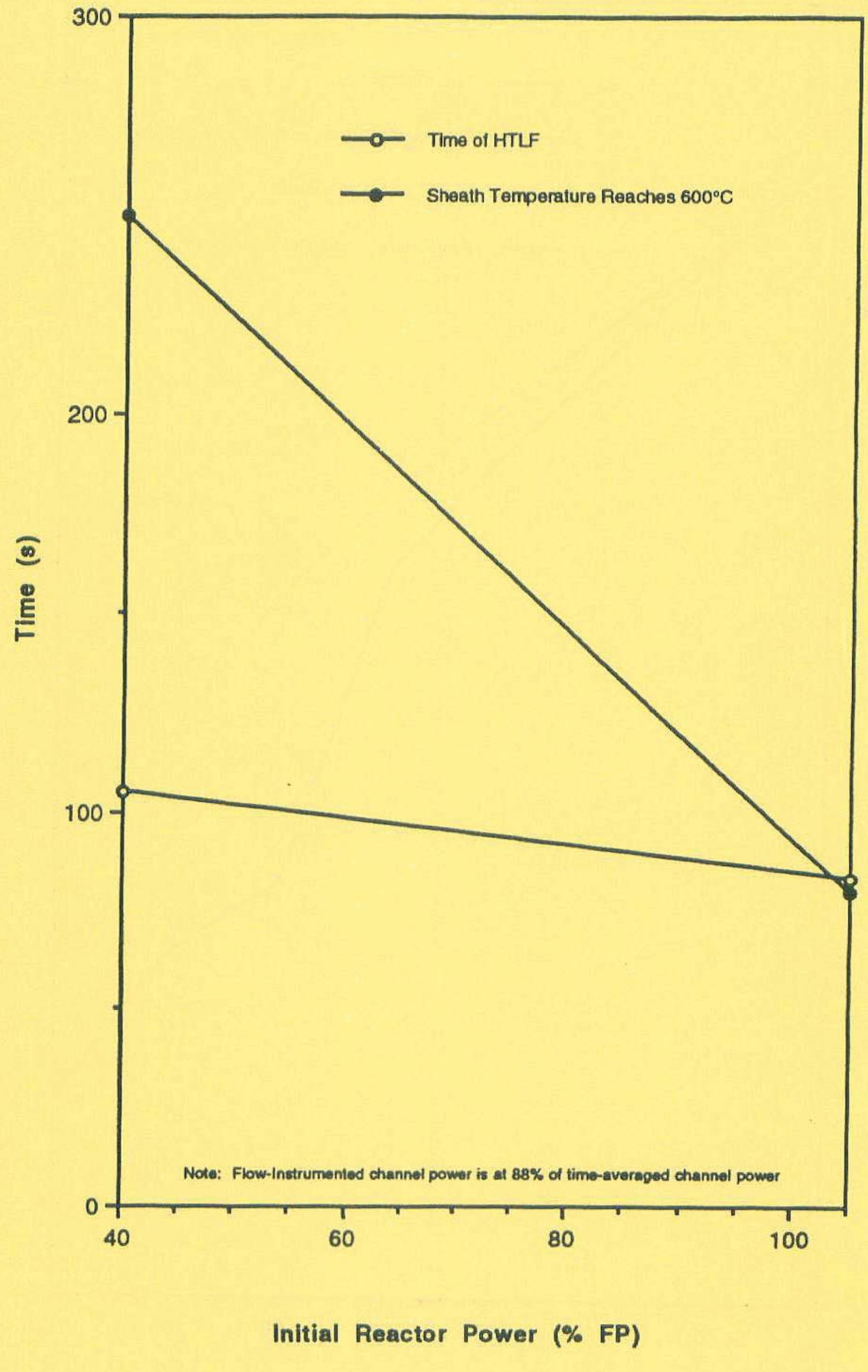


Figure 6(c)

Event Timing for Time of HTLF Trip and Time Sheath Temperature Reaches 600°C During 230 kg/s In-Core Break, With Moderator Level Control, Equilibrium Fuel Conditions, 0 wt% Isotopic Different, 18 mk Moderator Poison

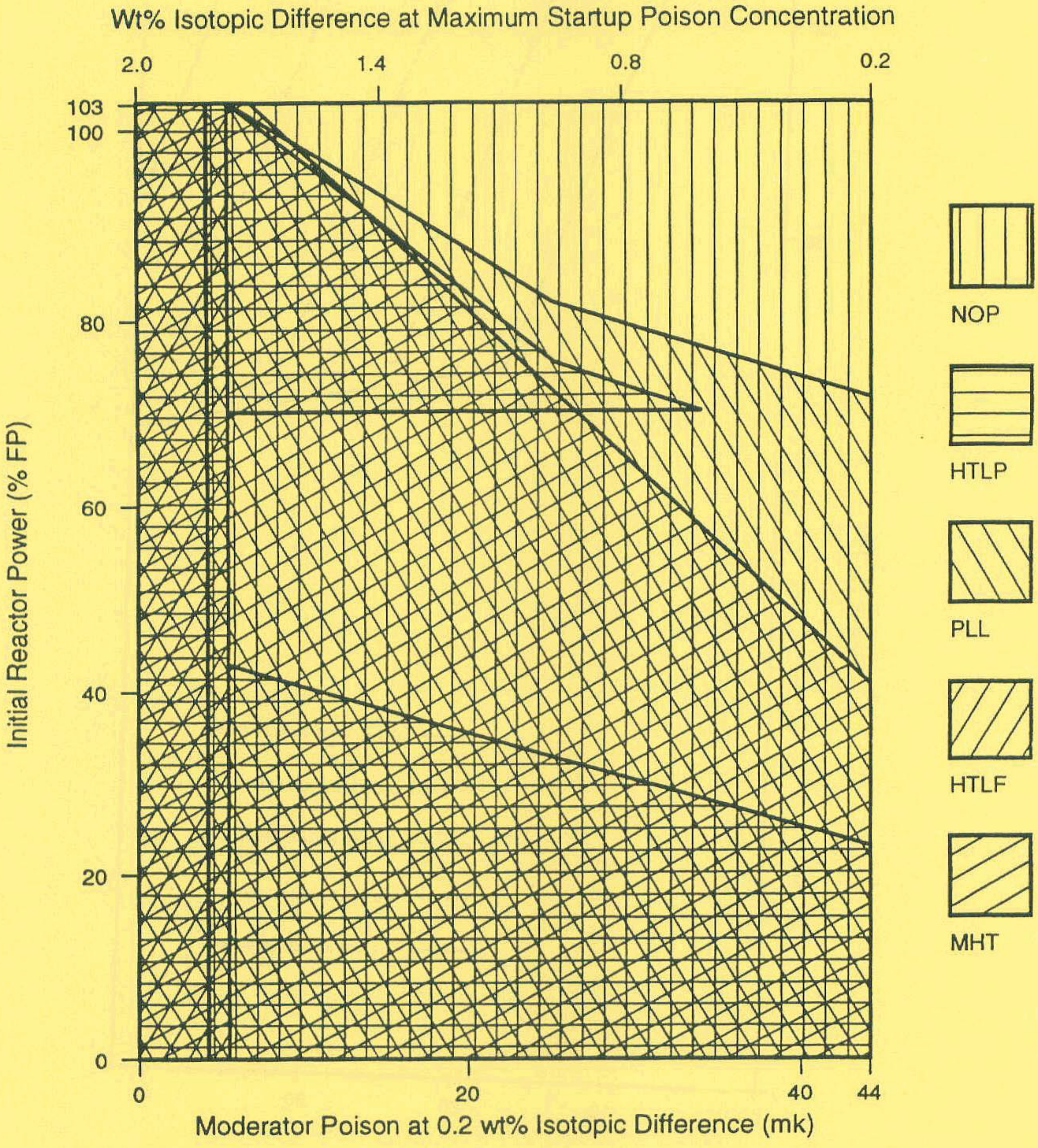


Figure 7

SDS1 Trip Parameter Effectiveness for
225 kg/s In-Core Break During 4 HT Pump Operation
(Bruce B)

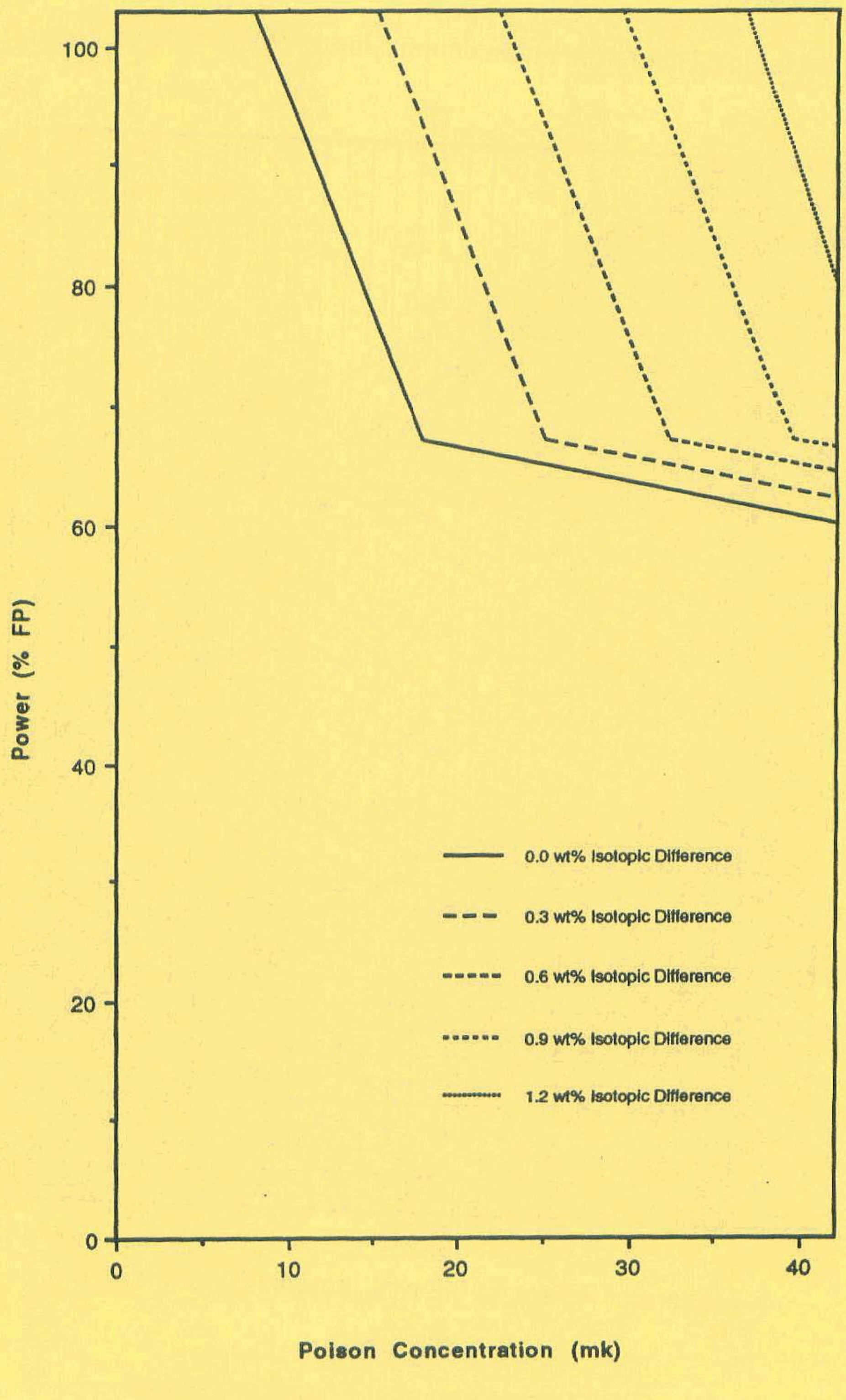


Figure 8

Pickering A Operating Limits