APPENDIX C

CATEGORY C
TRANSIENT FUEL ROD ANALYSIS

PHENOMENA DESCRIPTIONS AND RATIONALES FOR IMPORTANCE RANKING, APPLICABILITY, AND UNCERTAINTY

This appendix provides a description for each phenomenon appearing in Table 3-4, Transient Fuel Rod Analysis PIRT. Entries in the Table C-1, columns 1 and 2, follow the same order as in Table 3-3. Table C-1, column 3, also documents the PIRT-panel developed rationales for three types of Panel findings.

First, rationales are provided for the importance (High, Medium, or Low) assigned by the panel to each phenomenon. Because importance ranking was established by a vote of the panel members, a rationale is provided whenever one or more panel members voted a particular rank, i.e., High, Medium or Low. If there were no votes for a given importance rank, "No votes" is entered.

Second, the PIRT panel considered the applicability of the baseline PIRT to a broader set of circumstances, e.g., different fuel arrays, cladding types, reactor types, and burnups to 75 GWd/t. The specific question addressed by the PIRT panel was as follows: "Could the importance ranking assigned for the given phenomenon in the baseline PIRT be different for other fuel arrays, cladding types, reactor types, or burnups?" If this question is answered with a "no", the following entry appears in Table C-1: "Baseline PIRT importance rank is applicable." If this question is answered with a "yes", the rationale is entered. Additional details are presented in the footnotes to Table 3-4.

Third, the PIRT panel considered the current state of knowledge or uncertainty regarding each phenomenon. The phenomenon is characterized as "known (K)" if approximately 75-100% of full knowledge and understanding of the phenomenon exists. The phenomenon is characterized as "partially known (PK)" if between 25-75% of full knowledge and understanding of the phenomenon exists. The phenomenon is characterized as "unknown (UK)"if less than 25% of full knowledge and understanding of the phenomenon exists. Because the uncertainty ranking was established by a vote of the panel members, a rationale is provided whenever one or more panel members voted a particular uncertainty, i.e., known, partially known, or unknown. If there were no votes for a given uncertainty level, "No votes" is entered.

Table C-1. BWR Power Oscillations without Scram. Category C – Transient Fuel Rod Analysis

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, and Uncertainty)
Initial conditions	Gap size	The dimension (size) of the space between the pellet and cladding. H(0) No votes. M(0) No votes. L(6) The transient is occurring at different than initial conditions. The gap would be closed regardless of initial size. All: Baseline PIRT importance ranks applicable. K(3): Gap size calculations are sufficiently accurate. PK(1): Pellet cracking. UK(0): No votes.
Initial conditions	Gas pressure	The total pressure input to the code as the initial condition. H(0) No votes. M(0) No votes. L(7) At the time of melting, the gap has closed and the contact conductance is dominant. All: Baseline PIRT importance ranks applicable. K(7): Gas pressure calculation is sufficiently accurate, but uncertainty will increase with exposure. PK(0): No votes. UK(0): No votes.

Table C-1. BWR Power Oscillations without Scram. Category C – Transient Fuel Rod Analysis (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, and Uncertainty)
Initial conditions	Gas composition	The composition of the gas input as the initial condition. H(0) No votes. M(0) No votes. L(7) At the time of melting, the gap has closed and the contact conductance is dominant. All: Baseline PIRT importance ranks applicable. K(6): Gas pressure calculation is sufficiently accurate. PK(1): May not be accurately known for high exposure. UK(0): No votes.
Initial conditions	Gas distribution	The axial and radial distribution of the gas input as the initial condition (inter, intra, porosity). H(0) No votes. M(0) No votes. L(7) For this calculation, only the effect of fission gas on thermal conductivity is important. That effect is covered in another phenomenon. All: Baseline PIRT importance ranks applicable. K(1): Gas pressure and distribution are accurately known. PK(5): Not enough hot cell data for newer fuel types. UK(0): No votes.

Table C-1. BWR Power Oscillations without Scram. Category C – Transient Fuel Rod Analysis (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, and Uncertainty)
Initial conditions	Pellet and cladding dimensions	Characteristic physical dimensions. H(4) The initial dimensions set the geometry for the problem. They affect the heat transfer area and time constants. M(3) Same as high rationale but of less importance compared to other phenomena. L(0) No votes. All: Baseline PIRT importance ranks applicable. K(7): Pellets and cladding tubes are manufactured to tight tolerances. PK(0): No votes. UK(0): No votes.
Initial conditions	Burnup distribution	The radial and axial burnup magnitude and distribution in the fuel specified as the initial condition. H(5) Affects power distribution, conductivity, and melting temperature, all of which are important to melting. M(0) No votes. L(0) No votes. All: Baseline PIRT importance ranks applicable. K(6): This phenomenon is well known. PK(0): No votes. UK(0): No votes.

Table C-1. BWR Power Oscillations without Scram. Category C – Transient Fuel Rod Analysis (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, and Uncertainty)
Initial conditions	Hydrogen concentration	The average hydrogen concentration in the cladding specified as the initial condition. H(0) No votes. M(0) No votes. L(7) Doesn't affect the mechanism of fuel melting. All: Baseline PIRT importance ranks applicable. K(1): Can be calculated from first principles. PK(6): Uncertainties exist about boundary conditions, e.g., water chemistry. UK(0): No votes.
Initial conditions	Hydrogen distribution	The local distribution of hydrogen in the cladding and hydride orientation specified as the initial condition. H(0) No votes. M(0) No votes. L(7) Doesn't affect the mechanism of fuel melting. All: Baseline PIRT importance ranks applicable. K(1): Can be calculated from first principles. PK(6): Uncertainties exist about boundary conditions, e.g., water chemistry. UK(0): No votes.

Table C-1. BWR Power Oscillations without Scram. Category C – Transient Fuel Rod Analysis (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, and Uncertainty)
Initial conditions	Fast fluence	Time integrated fast neutron flux to which the cladding is exposed. H(0) No votes. M(0) No votes. L(5) By the time the fuel melts, radiation damage to the cladding has been annealed out. All: Baseline PIRT importance ranks applicable. K(6): Can be calculated accurately. PK(0): No votes. UK(0): No votes.
Initial conditions	Porosity distribution	The porosity distribution, including the rim, specified as the initial condition that is used to calculate the thermal conductivity and the fission gas transient behavior. H(0) No votes. M(5) Influences fuel centerline temperature for given conditions. L(0) No votes. All: Baseline PIRT importance ranks applicable. K(0): No votes. PK(6): Can be calculated but uncertainty is large. UK(0): No votes.

Table C-1. BWR Power Oscillations without Scram. Category C – Transient Fuel Rod Analysis (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, and Uncertainty)
Initial conditions	Rim size	Width of zone at outer periphery of pellet characterized by high porosity, high local burnup and plutonium content, and small grain structure containing fission gases in tiny closed pores specified as the initial condition. H(0) No votes. M(4) Has a similar thermal effect as the porosity distribution although the fuel temperatures being affected are nearer the periphery. L(2) Power distribution effect of plutonium in rim is what matters and that is accounted for in the power distribution phenomenon that follows. Fuel: Importance may be reduced for MOX fuel. Clad: Baseline PIRT importance rank applicable. Reactor: Baseline PIRT importance rank applicable. Burnup: Baseline PIRT importance rank applicable. K(0): No votes. PK(5): Can be predicted, but data has high uncertainty. UK(0): No votes.
Initial conditions	Power distribution	The radial and axial magnitude and distribution of the power produced within the fuel rod, including the effect of plutonium in the rim region. H(7) Influences stability characteristics and radial temperature distribution in the fuel rod. M(0) No votes. L(0) No votes. All: Baseline PIRT importance ranks applicable. K(6): Can calculate to within 25%. PK(0): No votes. UK(0): No votes.

Table C-1. BWR Power Oscillations without Scram. Category C – Transient Fuel Rod Analysis (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, and Uncertainty)
Initial conditions	Fuel-gap friction coefficient	The friction coefficient between the pellet and cladding specified as an initial condition to represent the initial-state interaction between the two. H(0) No votes. M(0) No votes. L(6) Primarily a mechanical effect that affects the stresses in the cladding at the locations of fuel cracking. This does not have a significant impact on the fuel behavior. All: Baseline PIRT importance ranks applicable. K(1): Experimental data are available and the friction factor is known within 25%. PK(1): Same as for known but the uncertainty is greater than 25%. UK(0): No votes.
Initial conditions	Thickness of oxide layer and surface condition (rewet)	The condition of the oxide layer specified as the initial condition, including thickness and surface characteristics. H(6) Surface condition affects the rewet temperature. Oxidized fuel is at lower risk of fuel melting because the oxide coating increases the probability of rewet. M(0) No votes. L(0) No votes. All: Baseline PIRT importance ranks applicable. K(0): No votes. PK(6): Can be calculated from first principles, but the uncertainty is greater than 25%. UK(0): No votes.

Table C-1. BWR Power Oscillations without Scram. Category C – Transient Fuel Rod Analysis (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, and Uncertainty)
Initial conditions	Rod free volume	The plenum and other free volumes within the fuel stack specified as an initial condition. H(0) No votes. M(0) No votes. L(7) No significant impact on fuel melting. All: Baseline PIRT importance ranks applicable. K(6): Fuel mass and volume are known to a high degree of accuracy. PK(0): No votes. UK(0): No votes.
Mechanical loading to cladding	Pellet thermal expansion including expansion due to fuel melting	The change in pellet dimensions induced by cyclic changes in the pellet temperature (the magnitude of the change is proportional to the material coefficient of thermal expansion) or by phase change (melting). H(6) Deformation of the pellet drives the cladding loading. M(1) Cladding is ductile when fuel starts to melt and can accommodate the stress. L(0) No votes. All: Baseline PIRT importance ranks applicable. K(0): No votes. PK(6): Pellet expansion is well known, but impact on cladding mechanical loading is less well known. UK(0): No votes.

Table C-1. BWR Power Oscillations without Scram. Category C – Transient Fuel Rod Analysis (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, and Uncertainty)
Mechanical loading to cladding	Direct gas pressure loading	The contribution of available fission gas combined with the fill gas in determining an internal pressurization. H(1) The release of fission gas will enhance the fuel temperature rise caused by the expanding gap. M(5) Less important than loading due to pellet expansion. L(0) No votes. All: Baseline PIRT importance ranks applicable. K(0): No votes. PK(6): Available volume to contain the released fission gases is not well known, because it varies with the degree of communication with the gas plenum. UK(0): No votes.
Mechanical loading to cladding	Pellet-cladding contact (gap closure)	The evolution of the pellet-cladding contact and associated friction coefficient evolution during the transient. H(6) Deformation of the pellet drives the cladding loading. M(1) Cladding is ductile when fuel starts to melt and can accommodate the stress. L(0) No votes. All: Baseline PIRT importance ranks applicable. K(0): No votes. PK(6): Consistent with and included in pellet thermal expansion. UK(0): No votes.

Table C-1. BWR Power Oscillations without Scram. Category C – Transient Fuel Rod Analysis (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, and Uncertainty)
Mechanical loading to cladding	Fission gas induced pellet swelling	The fission gas contribution to swelling of the pellet with the rapid increase in gas temperatures and pressure. H(4) Pellet swelling is enhanced at higher temperatures due to fission gas mobility, thus influencing cladding loading. M(4) Importance of this phenomenon somewhat less than thermal expansion. L(0) No votes. All: Baseline PIRT importance ranks applicable. K(0): No votes. PK(7): This phenomenon can be calculated, but there is a large uncertainty in translating the effect to mechanical loading. UK(0): No votes.
Mechanical loading to cladding	Fission gas release	The release of fission gas during the transient through the pellet into the gap. H(1) The release of fission gas will enhance the fuel temperature rise caused by the expanding gap. M(5) Less important than loading due to pellet expansion. L(0) No votes. All: Baseline PIRT importance ranks applicable. K(0): No votes. PK(6): This phenomenon can be calculated, but there is a large uncertainty in translating the effect to mechanical loading. UK(0): No votes.

Table C-1. BWR Power Oscillations without Scram. Category C – Transient Fuel Rod Analysis (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, and Uncertainty)
Fuel and cladding temperature changes	Heat resistances in fuel, gap, and cladding	The resistances offered by the fuel, gap, and cladding to the flow of thermal energy from regions of high temperature to regions of lower temperature. The resistance is dependent upon path length and thermal conductivity. H(8) Determines fuel temperature and heat flux at the cladding surface and influences, thereby, rewetting. M(0) No votes. L(0) No votes. All: Baseline PIRT importance ranks applicable. K(1): Fuel and clad thermal resistance are well known. PK(7): Fuel and clad thermal resistance are well known, but the uncertainty in the gap is more than 25%. UK(0): No votes.
Fuel and cladding temperature changes	Transient cladding-to-coolant heat transfer coefficient (oxidized cladding)	The correlation that determines transport of energy from the fuel rod to the coolant based on system calculations by one or more of the following modes: forced convection-liquid, nucleate boiling, transition boiling, film boiling, or forced convection-vapor, including transient critical heat flux and transient rewetting. H(8) Determines rewetting and whether this event progresses to fuel melting. M(0) No votes. L(0) No votes. All: Baseline PIRT importance ranks applicable. K(0): No votes. PK(8): This phenomenon is driven by key forcing phenomena. Uncertainty in correlations will not significantly change this ranking. UK(0): No votes.

Table C-1. BWR Power Oscillations without Scram. Category C – Transient Fuel Rod Analysis (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, and Uncertainty)
Fuel and cladding temperature changes	Heat capacities of fuel and cladding	The respective quantities of heat required to raise the fuel and cladding one degree in temperature at constant pressure. H(8) Governs the rate of temperature rise in the fuel. M(0) No votes. L(0) No votes. All: Baseline PIRT importance ranks applicable. K(8): This is the best known material property. PK(0): No votes. UK(0): No votes.

Table C-1. BWR Power Oscillations without Scram. Category C – Transient Fuel Rod Analysis (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, and Uncertainty)
Fuel and cladding temperature changes	Transient oxidation and energy source.	The change in cladding oxidation during the transient, including the energy source arising with cladding oxidation. H(5) Oxidation is a significant heat source at high cladding temperatures. M(3) Heat from oxidation is small relative to other heat sources. L(0) No votes. Fuel: Importance may be reduced for MOX fuel. Clad: Oxidation characteristics are material dependent and importance could change. Reactor: Baseline PIRT importance rank applicable. Burnup: Oxidation rate is dependent on the oxide thickness which is, in turn, burnup dependent and importance could change. K(0): No votes. PK(7): The nature of the phenomenon is well known and expressed in correlation. However, inputs to the correlations such as initial oxide thickness and temperature have large uncertainties. UK(0): No votes.

Table C-1. BWR Power Oscillations without Scram. Category C – Transient Fuel Rod Analysis (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, and Uncertainty)
Cladding deformation	Stress versus strain response	The change in the dimensions of the cladding due to the cumulative stresses imposed on the cladding as a result of the various loadings arising from the transients and the various factors inducing stress concentrations. H(5) Drives the size of the gap and, therefore, heat transfer. M(0) No votes. L(0) No votes. All: Baseline PIRT importance ranks applicable. K(6): Can be accurately calculated. PK(0): No votes. UK(0): No votes.
Cladding deformation	Strain rate effects	Strain rate effects as they change the stress strain curve in terms of affecting the yield stress and the deformation behavior in the plastic regime. H(0) No votes. M(0) No votes. L(6) Cladding is not operating in a regime where strain rate effects are important. All: Baseline PIRT importance ranks applicable. K(6): Material property is well known. PK(0): No votes. UK(0): No votes.

Table C-1. BWR Power Oscillations without Scram. Category C – Transient Fuel Rod Analysis (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, and Uncertainty)
Cladding deformation	Anisotropy	The variation of cladding properties along the different coordinate directions. H(0) No votes. M(0) No votes. L(6) Anisotropy disappears at high temperatures. All: Baseline PIRT importance ranks applicable. K(6): Material property is well known. PK(0): No votes. UK(0): No votes.
Cladding deformation	Pellet shape	Changes to the pellet shape from its initial state such as dished or champered ends, barreling or hourglassing as it affects the cladding response for ballooning. H(0) No votes. M(0) No votes. L(8) Not important considering the cladding loading mechanisms. Does influence fuel failure for this scenario. All: Baseline PIRT importance ranks applicable. K(0): No votes. PK(6): Only known well initially. UK(0): No votes.

Table C-1. BWR Power Oscillations without Scram. Category C – Transient Fuel Rod Analysis (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, and Uncertainty)
Cladding deformation	Cladding temperature	The effect of cladding temperature in determining cladding properties and leading to cladding deformation. H(6) Cladding temperatures can affect the pellet contact pressure and cladding deformation and heat transfer. M(0) No votes. L(0) No votes. All: Baseline PIRT importance ranks applicable. K(0): No votes. PK(6): Cladding temperature can be calculated to within 25% but the effect on cladding properties and eventually cladding deformation leads to higher uncertainties. UK(0): No votes.
Cladding deformation	Localized effects	Stress risers within the cladding at discrete locations arising from various sources, including the pellet shape factors listed above, as well as undetected defects in the cladding, as they affect the cladding circumferential temperature distribution. H(0) No votes. M(0) No votes. L(8) Not important for fuel melting failure mechanism. More important for ballooning. All: Baseline PIRT importance ranks applicable. K(0): No votes. PK(6): Not known very accurately. UK(0): No votes.

Table C-1. BWR Power Oscillations without Scram. Category C – Transient Fuel Rod Analysis (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, and Uncertainty)
Cladding deformation	Biaxiality	The dependence of cladding deformation and failure strain on the multidimensional stress state. H(0) No votes. M(0) No votes. L(8) Impact on cladding deformations is small and even smaller for fuel melting. All: Baseline PIRT importance ranks applicable. K(8): Can be calculated with reasonable accuracy. PK(0): No votes. UK(0): No votes.
Pellet deformation mechanisms	Fracture stress, yield stress in compression, plastic deformation, grain boundary decohesion, pellet cracking, and evolution of pellet stress state	Phenomena that change the pellet geometry and characteristics. H(4) Deformation of the pellet drives the cladding loading. M(1) Cladding is ductile when fuel starts to melt and can accommodate the stress. L(0) No votes. All: Baseline PIRT importance ranks applicable. K(1): No rationale provided. PK(6): The listed phenomena can be modeled, but the uncertainties in the models are large. UK(0): No votes.

Table C-1. BWR Power Oscillations without Scram. Category C – Transient Fuel Rod Analysis (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, and Uncertainty)
Forcing functions	Transient power distribution	The spatially dependent variation of power with time. H(8) This phenomenon determines the outcome of the event. M(0) No votes. L(0) No votes. All: Baseline PIRT importance ranks applicable. K(0): No votes. PK(8): Reasonably well known, but not within 25%. UK(0): No votes.
Forcing functions	Coolant conditions	The collection of coolant conditions making up the time varying coolant environment, e.g., coolant type, velocity, temperature, pressure, etc. H(8) This phenomenon drives the thermal power oscillations. M(0) No votes. L(0) No votes. All: Baseline PIRT importance ranks applicable. K(0): No votes. PK(8): Resonably well known, but not within 25%. UK(0): No votes.

Table C-1. BWR Power Oscillations without Scram. Category C – Transient Fuel Rod Analysis (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, and Uncertainty)
Multiple fuel rod and coolant channel interactions	Rod-to-rod interactions	Geometry changes in the bundle that lead to three-dimensional effects. H(3) Affects heat transfer and, therefore, pellet temperature and melting. Can have a significant impact on the outcome of the transient. M(4) Don't get significant fuel rod deformations (bowing) unless the event is well advanced. Therefore, this is a result of the event rather than a causative factor. L(1) Same as medium ranking rationale but even less importance assessed. Fuel: Importance may be affected by fuel and assembly design, e.g., 8x8 lattice versus 10x10. Clad: Baseline PIRT importance rank applicable. Reactor: Baseline PIRT importance rank applicable. Burnup: Baseline PIRT importance rank applicable. K(0): No votes. PK(1): When the rods will bow can be calculated but how the rods will bow cannot be accurately predicted. UK(6): Same as partially known but the uncertainty is sufficiently high that the phenomenon is characterized as unknown.

Table C-1. BWR Power Oscillations without Scram. Category C – Transient Fuel Rod Analysis (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, and Uncertainty)
Multiple fuel rod and coolant channel interactions	Rod-to-channel interactions	Geometry changes in the bundle that lead to three-dimensional effects. H(4) Could have a significant beneficial effect by enhancing cooling of the rods contacting the cladding. M(4) Same rationale as for the high but the influence is thought to be moderate. L(0) No votes. Fuel: Importance may be affected by fuel and assembly design, e.g., 8x8 or 10x10 lattice. Clad: Baseline PIRT importance rank applicable. Reactor: Baseline PIRT importance rank applicable. Burnup: Baseline PIRT importance rank applicable. K(0): No votes. PK(1): When the rods will bow can be calculated but how the rods will bow cannot be accurately predicted. UK(6): Same as partially known but the uncertainty is sufficiently high that the phenomenon is characterized as unknown.
Multiple fuel rod and coolant channel interactions	Rod and spacer grid interactions	Geometry changes in the bundle that lead to three-dimensional effects. H(7) Can cause enhanced cooling downstream of the spacers and force co-planar fuel failure further downstream. In addition, can cause rod bowing if axially expanding rods get trapped by the spacer. M(0) No votes. L(0) No votes. All: Baseline PIRT importance ranks applicable. K(5): Spacer effects are well known from full scale testing. PK(0): No votes. UK(0): No votes.

APPENDIX D

CATEGORY D
SEPARATE EFFECT TESTING

PHENOMENA DESCRIPTIONS AND RATIONALES FOR IMPORTANCE RANKING, APPLICABILITY, AND UNCERTAINTY

This appendix provides a description for each phenomenon appearing in Table 3-5, Separate Effect Testing PIRT. Entries in the Table D-1, columns 1 and 2, follow the same order as in Table 3-5. Table D-1, column 3, also documents the PIRT-panel developed rationales for three types of Panel findings.

First, rationales are provided for the importance (High, Medium, or Low) assigned by the panel to each phenomenon. Because importance ranking was established by a vote of the panel members, a rationale is provided whenever one or more panel members voted a particular rank, i.e., High, Medium or Low. If there were no votes for a given importance rank, "No votes" is entered.

Second, the PIRT panel considered the applicability of the baseline PIRT to a broader set of circumstances, e.g., different fuel arrays, cladding types, reactor types, and burnups to 75 GWd/t. The specific question addressed by the PIRT panel was as follows: "Could the importance ranking assigned for the given phenomenon in the baseline PIRT be for different for other fuel arrays, cladding types, reactor types, or burnups?" If this question is answered with a "no", the following entry appears in Table C-1: "Baseline PIRT importance rank is applicable." If this question is answered with a "yes", the rationale is entered. Additional details are presented in the footnotes to Table 3-5.

Third, the PIRT panel considered the current state of knowledge or uncertainty regarding each phenomenon. The panel determined that this area did not warrant further consideration (please see Section 3.4.4 of this report for the panel's reasons for this approach).

Table D-1. BWR Power Oscillations without Scram. Category D – Separate Effect Testing – Low Temperature

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, Uncertainty)
Specimen selection	Amount of oxide	The amount of zirconium oxide on both the inside and outside cladding surfaces. The oxygen source on the inner surface is UO_2 and the source on the outer surface is H_2O. H(0) No votes. M(7) Oxide thickness is not so important because over the normal range it is less than 10% of the cladding wall thickness likely to have little impact yield stress and elongation. L(0) No votes. All: Baseline PIRT applicable. K: Not applicable. PK: Not applicable. UK: Not applicable.
Specimen selection	Type of oxidation	Whether the cladding oxidation prior to testing was uniform and/or nodular. H(2) Presence of nodules may cause imperfections that are sufficient to localize the deformation under the nodule and thereby lower the strength of the cladding. M(5) Although there may be an effect, the real situation is little islands and the effect will be reduced for this situation. L(0) No votes. All: Baseline PIRT applicable. K: Not applicable. PK: Not applicable. UK: Not applicable.

Table D-1. BWR Power Oscillations without Scram. Category D – Separate Effect Testing – Low Temperature (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, Uncertainty)
Specimen selection	Cladding dimensions	The thickness and diameter of the fuel cladding. H(0) No votes. M(0) No votes. L(6) The mechanical properties are independent of cladding dimensions as long as the cladding dimensions are representative of claddings in use. All: Baseline PIRT applicable. K: Not applicable. PK: Not applicable. UK: Not applicable.
Specimen selection	Extent of oxide spalling and hydride blisters[a]	Peeling of the oxide layer from the cladding leaving the underlying material exposed to the coolant. Can lead to a local cold spot and hydride blister formation H(7) If oxide spalling is present it can lead to high local hydrogen distributions and reduce the overall ductility of the cladding. M(0) No votes. L(0) No votes. All: Baseline PIRT applicable. K: Not applicable. PK: Not applicable. UK: Not applicable.

[a] Discriminating factor: Rods that exhibit these characteristics should not be selected unless they occur to a significant extent in the population of rods to be investigated.

Table D-1. BWR Power Oscillations without Scram. Category D – Separate Effect Testing – Low Temperature (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, Uncertainty)
Specimen selection	Extent of oxide delamination	Separation of an outer oxide layer from the underlying oxide or base metal. Can lead to increased temperature and enhanced localized corrosion. H(0) No votes. M(2) If there is hydride redistribution in a hydrided region, it can affect mechanical properties. There is little possibility of this occurring in BWRs. Therefore, one should not select rods for testing if there are regions of delamination. L(0) No votes. All: Baseline PIRT applicable. K: Not applicable. PK: Not applicable. UK: Not applicable.
Specimen selection	Presence of barrier layer	Related to the presence or absence of barrier liner in the cladding. H(0) No votes. M(1) There is a potential for incipient defects from stress corrosion cracking. L(4) There are two considerations (1) lower strength of zirconium but since less than 10% of overall thickness and thus little effect, and (2) probability of incipient defect is expected to be low. All: Baseline PIRT applicable. K: Not applicable. PK: Not applicable. UK: Not applicable.

Table D-1. BWR Power Oscillations without Scram. Category D – Separate Effect Testing – Low Temperature (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, Uncertainty)
Specimen selection	Amount of hydrogen	Total amount of hydrogen in the cladding. H(0) No votes. M(6) Hydrogen can affect mechnical properties and the degree depends on the amount of hydrogen and distribution. Only a moderate amount of hydrogen is expected to be present and, therefore, the mechanical properties are only expected to be modestly affected. L(0) No votes. All: Baseline PIRT applicable. K: Not applicable. PK: Not applicable. UK: Not applicable.
Specimen selection	Hydrogen distribution	Spatial distribution of the hydrogen, including local hydride formations in the cladding (hydride rim) but excluding hydride blisters. H(0) No votes. M(6) Hydride rim could have a moderate impact on the outcome of the experiment. There is little choice regarding the hydrde rim; the phenomenon comes with the specimen. L(0) No votes. All: Baseline PIRT applicable. K: Not applicable. PK: Not applicable. UK: Not applicable.

Table D-1. BWR Power Oscillations without Scram. Category D – Separate Effect Testing – Low Temperature (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, Uncertainty)
Specimen selection	Hydride orientation[a]	The orientation of the hydrides, either axial or radial. H(4) If there are radial hydrides, they have significantly less ductility than the base metal and will significantly affect the abiltiy of the cladding wall to resist cladding deformation. M(2) Radial hydrides may have an effect on the properties but that depends on the amount of the hydrides and it is believed that the amount is modest. L(0) No votes. All: Baseline PIRT applicable. K: Not applicable. PK: Not applicable. UK: Not applicable.
Specimen selection	Fluence	Time-integrated particle flux to which the cladding is exposed. H(0) No votes. M(4) Precipitate dissolution does occur and does lead to the strengthening of the material and this effect should be characterized. L(2) Effects of fluence saturate out at low fluence levels and the impact on mechanical properties is captured at low fluence. All: Baseline PIRT applicable. K: Not applicable. PK: Not applicable. UK: Not applicable.

[a] Should determine the extent of radial hydriding and plan the testing program accordingly. Due to the more random texture of the BWR Zircaloy-2 cladding, radial hydrides are more likely than in PWR cladding.

Table D-1. BWR Power Oscillations without Scram. Category D – Separate Effect Testing – Low Temperature (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, Uncertainty)
Specimen selection	Cladding integrity	Whether the cladding is leak-proof, and whether it has any non-representative defects. H(7) Defects are bad; they are to be avoided in test specimens. M(0) No votes. L(0) No votes. All: Baseline PIRT applicable. K: Not applicable. PK: Not applicable. UK: Not applicable.
Test conditions	Test temperature	Identification and specification of a temperature at which the test is to be conducted (280-350 degrees C) and below where departure from nucleate boiling occurs H(0) No votes. M(0) No votes. L(6) The properties of the material are not likely to vary much within the specified range. All: Baseline PIRT applicable. K: Not applicable. PK: Not applicable. UK: Not applicable.

Table D-1. BWR Power Oscillations without Scram. Category D – Separate Effect Testing – Low Temperature (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, Uncertainty)
Test conditions	Strain rate	The specified rate of elongation imposed upon a test article. H(0) No votes. M(0) No votes. L(6) Given the evidence gathered for the RIA case that is likely to be bounding for this case, the effect of strain rate is small. All: Baseline PIRT applicable. K: Not applicable. PK: Not applicable. UK: Not applicable.
Test conditions	Stress state imposed on specimen	The type of stress that is applied to the material being tested. H(7) The ductility limit of Zircaloy cladding depends upon the cladding stress state. M(0) No votes. L(0) No votes. All: Baseline PIRT applicable. K: Not applicable. PK: Not applicable. UK: Not applicable.

Table D-1. BWR Power Oscillations without Scram. Category D – Separate Effect Testing – Low Temperature (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, Uncertainty)
Test conditions	Cycling conditions	Magnitude, duration, frequency and total number of strain cycles. H(4) Important to establish a record of whether there are path dependencies re ratcheting buildup or fatigue and to support the design of the integral tests. M(3) Based upon duration of event and limited number of cycles applied, the fatigue impact is expected to be modest. The imposed stress state is more important. L(0) No votes. All: Baseline PIRT applicable. K: Not applicable. PK: Not applicable. UK: Not applicable.
Test conditions	Tensile test specimen design	Design of the test specimen such that the appropriate, well-characterized stress state is invoked. H(7) It is important to ensure that the correct stress state has been achieved. M(0) No votes. L(0) No votes. All: Baseline PIRT applicable. K: Not applicable. PK: Not applicable. UK: Not applicable.

Table D-1. BWR Power Oscillations without Scram. Category D – Separate Effect Testing – Low Temperature (continued)

Subcategory	Phenomena	Definition and Rationale (Importance, Applicability, Uncertainty)
Test conditions	Burst specimen design	Design of the test specimen such that the appropriate, well-characterized stress state is invoked. When running a pressurized tube burst test, either with gas or oil, the stress state is such that there is twice as much stress in the hoop direction as in the axial direction. This factor is addressed in the design effort. H(7) It is important to ensure that the correct stress state has been achieved. M(0) No votes. L(0) No votes. All: Baseline PIRT applicable. K: Not applicable. PK: Not applicable. UK: Not applicable.

APPENDIX E

EXPERIMENTAL DATABASES

The experimental databases identified in Section 4 of this report are further discussed in this appendix. The author of each contribution is identified. The contributed documentation exhibits some style differences. References providing additional details for each test program are provided at the end of each contributed entry.

E-1. Separate Effect Tests

E-1.1. Cladding Mechanical Properties Tests (United States)

The information regarding this test series was provided by panel member A. Motta of the Pennsylvania State University and M. Billone of Argonne National Laboratory.

Argonne National Laboratory (ANL) and the Pennsylvania State University (PSU) are working together on a NRC-funded program to investigate cladding properties and to test loss-of-coolant accident (LOCA) acceptance criteria at high burnups. Although the main focus of the program is to investigate fuel behavior under LOCA conditions, related mechanical properties testing is being done under both LOCA conditions and RIA conditions. The tests at relatively low temperatures and high strain rates appropriate for RIA conditions are described briefly here. The objectives are two-fold: to understand the degradation in cladding failure behavior at high burnup and to obtain stress-strain relationships that will serve as inputs to codes. High-burnup fuel rods (about 70 GWd/MTU) from the H. B. Robinson PWR are expected to be available for these tests along with related archive fresh tubing. Although the fuel has not arrived at the time of this writing, high-burnup specimens (about 50 GWd/MTU) from TMI-1 are available and have been used for preliminary testing along with nonirradiated Zircaloy-4 tubing.

Ring-Stretch Tests. A ring tensile specimen design has been developed and tested at ANL to generate tensile properties in the hoop direction.[1] A related ring specimen design was developed and tested at PSU to provide a near plane-strain stress state that approximates the stress state produced by expanding fuel pellets during an RIA.[2,3] Tensile testing of cladding samples from archival tubing and high burnup rods will be performed over a temperature range from room temperature to 800 °C with strain rates from 0.1%/sec to 100%/sec on irradiated and nonirradiated specimens. Because hydrogen is expected to play an important role on the mechanical properties of the irradiated material, testing is also being done by PSU on artificially hydrided specimens of nonirradiated materials. These artificially hydrided samples allow us to investigate not only hydrogen content, but hydrogen distribution, i.e., when concentrated in a hydride rim or in blisters. Stress-strain relationships, along with tensile strengths (yield and ultimate) and elongations (uniform, total and local) will be measured as a function of temperature, strain rate, radiation damage, hydrogen, and oxygen content.

Axial Tensile Tests. Similar testing will be done on axial tensile specimens electromachined from de-fueled portions of irradiated fuel rods and from nonirradiated tubing specimens. These tests will be performed over the same temperature range and strain-rate range as the ring-stretch tests mentioned above. The combination of the axial and the hoop stress-strain properties will allow validation and improvement of the models used in fuel rod codes for predicting the mechanical behavior of an anisotropic alloy such as Zircaloy.

Biaxial Tube Burst Tests. Biaxial tube burst tests are the most informative and the most difficult to perform, and they consume the largest amount of specimen material, which is a significant consideration when testing irradiated fuel material. These tests will be done in a more limited temperature range of 300 °C -400°C, but they will explore the effects on deformation and failure of stress biaxiality ratios from 1:1 to 2:1 at high strain rate. In principle, the tests can be run with the fuel intact or with the fuel removed. Some tests will be run with the fuel removed to generate baseline data for code validation along with data that can be compared to other such studies on nonirradiated and medium-burnup cladding.

References for Cladding Material Properties Tests

1. A. B. Cohen et al., "Modified Ring Stretch Tensile Testing of Zr-1Nb Cladding," Proc. USNRC Water Reactor Safety Information Meeting, NUREG/CP-0162 **2**, 133-149 (October 20-22, 1977).
2. T.M. Link, D.A. Koss and A.T. Motta, "Failure of Zircaloy Cladding under Transvers Plane-strain Deformation," *Nuclear Engineering Design* **186** (1998) 379-394.
3. D. W. Bates, et al., "Influence of Specimen Design on the Deformation and Failure of Zircaloy Cladding," *Proc. ANS International Meeting on Light Water Reactor Fuel Performance,* Park City, Utah, 1201–1210 (April 10–13, 2000).

E-1.2. Cladding Mechanical Property Tests (Japan)

Ductility reduction due to hydrogen absorption and neutron irradiation was investigated for BWR cladding many years ago using the uniaxial tensile test, though both the hydrogen concentration and neutron fluence were much lower than the level currently of interest for high burnup fuels. Except for the general post-irradiation examination, BWR cladding has not been tested in recent years. Less significant corrosion and hydrogen pick-up than occurs in high burnup PWR fuel are important factors is this situation. However, ductility reduction in BWR cladding is possible in the expected high-burnup range. In the future, mechanical property tests for the cladding will be performed. The Japan Atomic Energy Research Institute (JAERI) is interested in the morphology and distribution of hydrides specific to the BWR cladding. Tube burst tests for hydrided claddings are planned.

E-2. Integral Tests

E-2.1. NSRR Pulse-Irradiation Experiments with BWR Fuels (Japan)

The information regarding this test series was provided by panel-member T. Fuketa.

The JAERI Nuclear Safety Research Reactor (NSRR) is a modified Training, Research, Isotopes, General Atomics-Annular-Core Pulse Reactor (TRIGA-ACPR) featuring a large pulsing power capability and large dry irradiation space located in the center of the reactor core. The experimental capsule used for the irradiated fuel rod test is a double-container system. The capsule contains an instrumented test fuel rod with stagnant water at atmospheric pressure and ambient temperature. The data obtained during the pulse irradiation includes cladding surface temperature, water coolant temperature, axial pellet stack and cladding tube elongations, fuel rod internal pressure, and capsule internal pressure. A water column velocity sensor is installed in some experiments for measurement of mechanical energy generation. A new capsule for high-temperature and high-pressure conditions is under development.

A total of twelve tests with irradiated BWR fuels were conducted in the NSRR by May 2000 (Table 3-1). The first series of five tests, TS, used 7x7 type fuel at burnup of 26 MWd/kgU from Tsuruga plant unit 1 reactor. In the tests, larger cladding deformation due to pellet cladding mechanical interaction (PCMI) than that of fresh fuel and considerable fission gas release of about 10% (to the production) were observed. The deformation was elastic and fuel failure did not occur at peak fuel enthalpies up to 410 J/g (98 cal/g). Cladding surface temperature remained around 100 °C.

The second series of tests, FK-1, -2 and -3, were conducted with 8x8 Step I type rods with Zr-liner cladding at burnup of 41 to 45 MWd/kgU from 1st Fukushima plant unit 3 reactor. In tests FK-1 and 3 at peak fuel enthalpies above 540 J/g, the cladding surface temperature reached about 600 °C. The threshold enthalpy of DNB appears to be higher than that of PWR fuels. The lower cladding temperature in the BWR fuel tests was likely caused by the wider pellet/cladding gap due to the smaller cladding creep down. A similar tendency is seen in residual hoop strain, suggesting that majority of the plastic strain occurred after the cladding temperature escalation.

The third series of tests, FK-4 and -5, used 8x8 step II type rods with Zr-liner cladding at burnup of 56 MWd/kgU irradiated for 4 cycles in 2nd Fukushima plant unit 2 reactor. The Step II fuel rod has narrower pellet/cladding gap and higher fuel density than those in the Step I rod. The general behavior of the Step-II rod, however, was quite similar to that of the Step I rod. The cladding condition in terms of oxide thickness and hydrogen content was similar to that of the Step I rod, i.e. about 20-μm thick oxide and 60 ppm hydrogen. The cladding was ductile enough to survive PCMI loading during the pulse irradiation in tests FK-1 through -5. With the larger fission gas release during the base irradiation, larger additional gas release during the pulse was observed in tests FK-4 and -5. This result suggests that the fission gas release during pulse correlates not only to the peak fuel enthalpy but also to the steady state gas release.

Table E-1. Irradiated BWR fuel tests in the NSRR

Test ID	Test Fuel	Fuel Burnup (MWd/kgU)	Peak Enthalpy (J/g)	Result
TS-1	7x7, 5 cycles Stress relieved cladding	26	230	No failure
TS-2	7x7, 5 cycles Stress relieved cladding	26	275	No failure
TS-3	7x7, 5 cycles Stress relieved cladding	26	366	No failure
TS-4	7x7, 5 cycles Stress relieved cladding	26	370	No failure
TS-5	7x7, 5 cycles Stress relieved cladding	26	410	No failure
FK-1	8x8, Step I, 5 cycles Zr-liner	45	544	No failure
FK-2	8x8, Step I, 5 cycles Zr-liner	45	293	No failure
FK-3	8x8, Step I, 5 cycles Zr-liner	41	607	No failure
FK-4	8x8, Step II, 4 cycles Zr-liner	56	586	No failure
FK-5	8x8, Step II, 4 cycles Zr-liner	56	293	No failure
FK-6	8x8, Step II, 5 cycles Zr-liner	61	548	Failed at 293 J/g, 100% fuel dispersed
FK-7	8x8, Step II, 5 cycles Zr-liner	61	540	Failed at 260 J/g, 100% fuel dispersed

The most recent tests FK-6 and -7 were conducted with fuels similar to those in tests FK-4 and -5 but at higher burnup of 61 MWd/kgU with 5 cycles irradiation. The fuel failed during the pulse irradiation at fuel enthalpies of 293 J/g (70 cal/g) and 260 J/g (62 cal/g) in tests FK-6 and -7, respectively. The expected peak fuel enthalpies were 548 J/g (131 cal/g) and 540 J/g (129 cal/g) in the two tests. The main difference in the two test conditions was the initial rod internal pressure, i.e. 0.1 MPa and 1.5 MPa (simulating EOL gas pressure) in tests FK-6 and 7, respectively. The cladding was separated into 3 pieces in both tests. All the fuel fragmented and dispersed into the capsule water. Post-test fuel examination and data analyses are in progress. Although the data from the tests FK-6 and 7 are preliminary, the results suggest occurrence of burst-type failure before escalation of cladding temperature.

References

1. T. Nakamura, et al., "Boiling Water Reactor Fuel Behavior at Burnup of 26GWd/tonne U under Reactivity-Initiated Accident Conditions," *Nuclear Technology* **108**, 45-60, (1994).

2. Fuketa, T., et al., "Behavior of PWR and BWR Fuels During Reactivity-Initiated Accident Conditions", *Proc. ANS International Meeting on Light Water Reactor Fuel Performance,* Park City, Utah, CD-ROM, (April 10-13, 2000).

1. Nakamura, T., et al., "Boiling Water Reactor Fuel Behavior under Reactivity-Initiated- Accident Conditions at Burnup of 41 to 45GWd/tonne U," *Nuclear Technology* **129**, (2000).

E-2.2. SPERT Test Reactor Data (United States)

The information regarding this test series was provided by R. Meyer, US NRC.

There were several Special Power Excursion Reactor Test (SPERT) facilities with different reactor cores; the core used for the tests of interest was the Capsule Driver Core (CDC); these tests are often referred to as the CDC tests. Single rods were tested in an instrumented water-filled capsule at ambient conditions. SPERT with the CDC had a natural pulse width of about 20 ms.

Table E-2 lists the characteristics of the irradiated fuel tests in the SPERT reactor. The test rods were BWR-type fuel rods manufactured to specifications being used by GE at that time except that many of the rods had smaller outside diameter to achieve higher energy depositions. All the rods listed in the following table are of the smaller size except CDC-703 and -709. These smaller rods also had a correspondingly reduced cladding thickness and gas gap. Preirradiation to accumulate the burnup was done in the Engineering Test Reactor at typically high power levels.

It is interesting to note that test CDC-859, which exhibited a low energy failure, was widely believed to be invalid because of presumed prior cladding failure and waterlogging. That belief was not correct, however, and it is now recognized that CDC-859 shows the typical low energy, brittle fracture characteristic of high-burnup Zircaloy rods having significant cladding oxidation. CDC-756 exhibited similar characteristics, although the failure energy level was higher.

Table E-2. Characteristics of BWR-Type Specimens Tested in Stagnant Water at an Initial Temperature of 20°C in the SPERT Test Reactor

Test No.	Burnup (GWd/t)	Oxide Thick. (μ)	Pulse Width (ms)	Peak Fuel Enthalpy (cal/g)	Clad. Fail (Yes/No)	Comments
CDC-571	4.6	~0	31	134	No	
CDC-568	3.5	~0	24	165	Yes	Enthalpy at failure 147 cal/g
CDC-567	3.1	~0	18	219	Yes	Enthalpy at failure 214 cal/g
CDC-569	4.1	~0	14	289	Yes	Enthalpy at failure 282
CDC-703	1.1	~0	15	159	No	12% diametral swelling
CDC-709	1	~0	13	198	Yes	Enthalpy at failure 190 cal/g; rod failed 280 ms after peak power
CDC-685	13.1	~0	23	154	No	
CDC-684	12.9	~0	20	166	No	Slight swelling
CDC-756	32.7	65	17	146	Yes	Enthalpy at failure <143 cal/g; one tiny PCMI crack
CDC-859	31.8	65	16	158	Yes	Enthalpy at failure 85 cal/g; very little fuel loss; three large PCMI cracks

E-2.3. Transient Critical Heat Flux Experiments and Rewet Data

The information presented in this section was provided by the following individuals: J. G. M. Anderson of Global Nuclear Fuels, B. Dunn of Framatome Technologies, L. E. Hochreiter of the Pennsylvania State University, J. Leikov of ASEA/ABB/Westinghouse Atom, D. W. Pruitt of Siemens Power Corporation, and J. Tulenko of the University of Florida. All, with the exception of J. Leikov, are members of the PIRT panel.

Introduction. During a PWR or BWR LOCA, the fuel will experience an early boiling transition (critical heat flux) during the flow transient immediately following the break. In a PWR LOCA, core flow reversals will provide improved cooling but the hot rod will not be sufficiently cooled such that it rewets. The high power PWR rods will remain above the rewet temperature until the reflood period of the transient. In a BWR LOCA, the early boiling transition is terminated by lower plenum flashing which causes a transient core flow increase, which can rewet the cladding. Later in the BWR LOCA the fuel will experience a second boiling transition due to loss of liquid inventory within the vessel, and a cladding heat up occurs. For both the PWR and BWR LOCA, the cladding heatup is terminated by reflooding of the reactor vessel by the plant Emergency Core Cooling Systems (ECCS). The prediction of transient boiling transition (critical heat flux) and rewet is essential for the evaluation of the fuel performance for these events.

The power oscillation without scram accident and the LOCA have been identified as key events for the evaluation of fuel performance for a BWR. In the power oscillation without scram accident, the BWR will be at a low flow natural circulation state and can experience flow oscillations which result in power oscillations. During these oscillations the high power fuel bundles may undergo periodic boiling transition (critical heat flux) and rewet following each power pulse as the flow pulses into and out of the reactor core. As long as the linear heat rate is sufficiently low, the cladding temperature remains below the minimum film boiling temperature (Tmin) during the oscillations, the cladding will rewet as the flow enters the core and excessive fuel heat up is avoided. However, if the fuel linear heat rate is sufficiently high, the cladding temperature can exceed the minimum film boiling temperature (approximately 600 °C (1100 °F)) following a power pulse, the cladding may not rewet such that substantial fuel heat up may occur.

At the April 2000 NRC-sponsored BWR ATWS PIRT for High Burnup Fuel meeting, it was decided that the available data for transient boiling transition (critical heat flux) and rewet would be examined and summarized for both BWR and PWR situations. The questions to be addressed were; if the use of steady-state critical heat flux correlations was sufficiently validated and accurate for transient situations, and if sufficient data existed such that an accurate assessment of a suitable minimum film boiling temperature (Tmin) could be used. This summary report addresses these two questions.

BWR Transient Dryout and Rewet Tests. Data for transient dryout, post dryout heat transfer and transient rewet have been obtained since the 1960's. The data include simple geometry tests as well as full scale simulated fuel bundles.

Simple geometry data [1,2,3] have typically been obtained in tubular and annular geometries and include steady state as well as transient tests. These tests typically give well-defined thermal hydraulic data and are excellent for model qualification. They do not, however, provide information on the cross sectional variation of thermal hydraulic conditions in a rod bundle. The maximum peak cladding temperature (PCT) for these experiments goes well beyond the minimum film boiling temperature, where rewet is not obtained. These tests therefore provide valuable information on boiling transition, film boiling heat transfer and rewetting.

Similar tests have been obtained in simple rod bundles [4,5,14], typically 4x4 rod bundles. In these tests both steady state and transient tests have been performed. The steady state test were used to obtain information on film boiling heat transfer, while the transient tests were used to obtain additional information on transient dryout and rewet. The transients were either simple power and flow transients where either the power was temporarily raised or the flow temporarily reduced to obtain a boiling transition, or they were simulation of a reactor turbine trip or recirculation pump trip. These tests also give PCTs beyond the minimum film boiling temperature and provide valuable information on boiling transition, film boiling heat transfer and rewetting.

BWR fuel vendors perform extensive critical power tests for each new fuel product that is developed. Steady state critical power data over a range of parameters covering normal steady state operation as well as the expected range of parameters for operational transients. These data are used to develop a fuel type specific critical power correlation. In addition a few transient tests are usually performed to demonstrate the applicability of the correlation under transient conditions [6,7,8,11,19,27,28,29,30,31]. The tests include simulated turbine trip and recirculation pump trip transients, and in one case reactor instability was simulated. Since the transient tests are intended to demonstrate the applicability of the critical power correlation under transient conditions, the PCT typically does not exceed the saturation temperature by more than 100-200 °C and thus does not provide data beyond the minimum film boiling temperature.

ASEA-ATOM has also performed FIX-II experiments resulting in high transient post dryout temperatures. These tests were performed for older fuel assembly designs, however they are well documented. The tests consisted of quick and slow flow decreases and step change power increases. The tests were performed on a full length 6x6 rod array and had measured cladding temperatures of up to 700 °C. ASEA also performed transient experiments for the current 10x10 fuel assembly designs including transient dryout as well as rewet. There are approximately 400 tests for different fuel assembly designs with different spacer grid designs and number of spacers. The tests also include experiments with fast changes in power and flow and a combination of different fluid conditions.

ASEA-ATOM also performed stability measurements for the current fuel assembly designs including flow oscillations, that for some cases resulted in a diverging situations which gave alternating oscillating dryout and rewet behavior. These tests were performed at low system pressures of approximately 16 to 30 bars (230 to 450 psia). ASEA-ATOM has also performed post-dryout, film boiling experiments that can be used to evaluate rod bundle film boiling heat transfer coefficients.

Siemens Power corporation has also performed stability measurements for its current fuel assembly designs in the Karlstein test facility. These tests were conducted at prototypic operating conditions using full size bundles. The tests measured the bundle stability characteristics up to the instability threshold. In some of the experiments, the power was maintained and increased to intentionally observe boiling transition and rewetting under oscillatory conditions (44).

Numerous Loss Of Coolant Accident (LOCA) experiments have provided information on transient dryout, film boiling heat transfer and transient rewet. These tests include data from the BDHT [15], TLTA [16], FIST [17,18], FIX [9,10,21,22,23], TBL [24] and ROSA-111 [25,26] test facilities which are all scaled simulation of a boiling water reactor (BWR). The upper range of PCTs for these tests is approximately 870 °C (1600 °F). High temperature data up to 1150 °C (2100 °F) have been obtained in GE's core spray heat transfer test facility and from the GOTA test facility and similar facilities at Hitachi and Toshiba [20].

Finally in-pile experiments have been performed, where nuclear fuel rods have been subject to boiling transition during power and flow transients. Even though the primary purpose of these tests was to evaluate the thermal/mechanical response of the fuel, these tests also provide valuable data on transient dryout and rewet. The early data in the Van Houten report [12] were collected for exposures up to 20GWd/t and PCTs up to 1700C. The later data from the Halden test reactor [13] had exposures up to 40GWd/t and PCTs up to 950 °C. BWR transient dryout and- rewet tests are summarized in the following table.

A recent paper (32) has examined density wave oscillations in the Japanese Advanced Thermal Reactor in which the cladding experienced dryout and rewetting. The channel powers were increased until continued dryout occurred. The calculations could reproduce the trends of the experiments but not the detail of the oscillations.

PWR Transient Critical Heat Flux and Rewet Experiments. There have also been many transient critical heat flux (or DNB) experiments to examine the possibility of a delay in critical heat flux in the initial stages of a PWR LOCA (Table E-3). A delay in critical heat flux would allow for the removal of a significant amount of the initial stored energy in the fuel rod such that the resulting fuel heatup after critical heat flux would be much less severe. Much of the work began in the late 1960's and continued through the early 1980's. The early experiments were either tube

Table E-3. Transient Dryout and Rewet Tests.

Geometry	Test Type	PCT	References
Simple Geometry Tests			
Tubular and Annular	Steady State and Transient	850 °C	1, 2, 3
Simple Rod Bundles			
4X4 Rod Bundles	Steady State Film Boiling Flow and Power Transients Simulated Turbine and Pump Trips	715 °C	4, 5, 14
Full Scale Rod Bundles			
	Simulated Turbine and Pump Trips for 8X8, 9X9 and 10X10 Rod Bundles	510 °C	6, 7, 8, 11, 19, 27, 28, 29, 30, 31
LOCA			
Scaled Simulation of a BWR.	BDHT, TLTA, FIST, FIX, TBL, ROSA	870 °C	9, 10, 15, 16, 17, 18, 21, 22, 23, 24, 25, 26
Core Spray Heat Transfer	CSHT, GOTA, Toshiba, Hitachi	1150 °C	20
In-Pile Data			
	Flow and Power Transients oscillations	1700 °C	12, 13

Note: Minimum Film Boiling Temperature ~ 600 °C.

experiments or smaller rod bundle experiments which were not full length. Some of the initial tests were performed by AEROJET on the original Semi-scale test facility and showed a delay in the critical heat flux of several seconds (33) instead of an almost instantaneous critical heat flux during blowdown. It is believed that these early tests were hot leg break simulations. Early experiments performed by ORNL for a hot leg break simulation (34) also showed a delay in critical heat flux of several seconds during blowdown. For both simulations, since the break is in the hot leg no flow stagnation would occur. For this type of break, the coolant inventory would have to pass through the core before it exited the break and would provide good cooling which would delay CHF. The Semi-scale and ORNL experiments were analyzed by Babcock and Wilcox as part of their direct testimony at the 1972 Core Cooling Hearings (35) using their CRAFT code. B&W found that if they suppressed critical heat flux (or DNB) until the fluid quality was calculated to be unity (quality = 1), they achieved reasonable agreement with the experimental data. This is logical for a hot leg break since good cooling will occur until the inventory is depleted. B&W constructed a 9 rod, one-half length full pressure test facility and performed additional transient DNB experiments including both hot leg and cold leg break simulations (36). Experiments were performed for a single ended cold leg break as well as for a double-ended cold leg break. Calculations with their CRAFT code with CHF suppressed until the calculated flow quality was unity agreed with the test data. These calculations indicated a significant delay in CHF and support that the use of steady-state CHF correlations for a transient situation was conservative.

Other experimental studies on transient critical heat flux are summarized in J.C.M. Leung in ANL-78-39 (37). This report gives 153 references of different experiments and analyses in which steady-state critical heat flux correlations were compared to the transient CHF data. Nearly all the experiments were small scale tubes, annuli, or small bundles which were less than full length. There were some full-length tube data but no full-length rod bundle data. Comparisons of steady-state CHF correlations with the test data from these experiments indicated that the steady-state correlations would predict earlier critical heat flux as compared to the experiments, that is, the correlations would conservatively predict that critical heat flux would occur before it was observed to occur in the experiments.

Westinghouse performed transient critical heat flux experiments on a 5x5 bundle which was full length (38, 39, 40). The experimental facility could represent full PWR operating pressure, temperatures and mass flux operating conditions. The heater rods could also represent the full power capabilities of operating PWR fuel rods for steady and transient conditions. There were two types of experiments performed; single parameter experiments in which one parameter, flow decay or pressure decay, was changed at a time, and system response experiments in which the test facility could represent the initial stages of a PWR large-break-LOCA with flow reversals and stagnation flow to simulate different break sizes. The data was analyzed on a bundle average basis to obtain the fluid conditions as well as on a subchannel basis.

Several different critical heat flux correlations were compared to the experimental data. Two criteria were used when comparing the correlations; the timing of the initial critical heat flux occurrence and the extent of CHF along the rod bundle. For the controlled parameter experiments, the W-3 correlation predicted CHF earlier than the measured time, as did the B&W-2 correlation. The B&W-2 correlation gave better predictions than the other correlations. The MacBeth correlation failed to predict CHF in a number of tests when it was observed and the Biasi correlation tended to predict longer times to CHF as compared to the data. In most cases the axial extent of CHF was under predicted, by all correlations. One point to note was that the CHF did occur at qualities that were significantly lower than unity which is different than that observed in the earlier B&W experiments.

For the natural blowdown experiments which simulated a double-ended LOCA, the extent of CHF in the rod bundle was very large. It should also be noted that the bundle heater rod measurements did indicate that CHF occurred very rapidly at nearly all elevations with CHF being observed between 0.5 and 1 seconds. The W-3 correlation tended to predict a longer delay in CHF than was observed, while the B&W-2 and Biasi correlation agreed very well with the data for both timing and extent of CHF. The MacBeth correlation was mixed, with earlier predictions at the bottom 1/3` of the bundle and late predictions for the top 2/3 of the bundle. The calculated fluid conditions at the time of CHF indicated qualities in the range of 0.4 to 0.75 % with corresponding void fraction of 0.85 to 0.97 indicating annular flow.

Rewetting has been studied extensively as part of PWR reflood heat transfer for the last 30 years. The most significant reflood experiments have been summarized in

The Rod Bundle Heat Transfer Program Report (41). For low pressure situations, a rewet temperature of 900 - 1000 ^{0} F is reasonable for stainless steel or Inconel cladding electrical heater rods. Reflood data on Zircaloy clad rods does indicate higher rewet temperatures. Also, if oxidation layers exist on the cladding, the rewet temperature can also increase.

Rewetting during blowdown has also been studied using stainless steel electrical heater rods as well as nuclear fuel rods. Rewetting temperatures have been published for blowdown situations for electrical heater rods (42), and a distribution of the measured rewet temperatures was given as seen in Figure E-1. As this figure indicates, a rewet temperature of approximately 900 to 1100 °F is reasonable. Rewet of the LOFT Zircaloy clad rods also occurred during blowdown at high temperatures, approximately 1300 °F for test L2-6 (43). It is believed that the LOFT blowdown quench temperatures are under estimated due to the presence of the external thermocouples, that is, the real surface temperatures of the LOFT rods were higher than measured.

Conclusions. The questions which were posed in the Introduction were:

The first question posed was: "Is the use of steady-state critical heat flux correlations sufficiently validated and accurate for transient conditions?"

Based on the information presented in this report and listed in the references, there is a sufficient database and validation for the use of steady-state critical heat flux correlations for transient CHF situations. In most cases, the use of steady-state critical heat flux correlations tended to predict CHF before it was observed to occur in the experiments. The scatter in the delayed CHF data is not well understood and until it is, a conservative approach must be used.

The second question that was posed was: "does sufficient data exist such that an accurate assessment of a suitable minimum film boiling temperature could be used?"

Based on the information presented in this report there is sufficient Tmin data which is available which can be used to establish a suitable value which can be used in confidence. Data does exist for both blowdown and reflood situations. There is significant scatter of the data that should be considered in establishing a specific Tmin value.

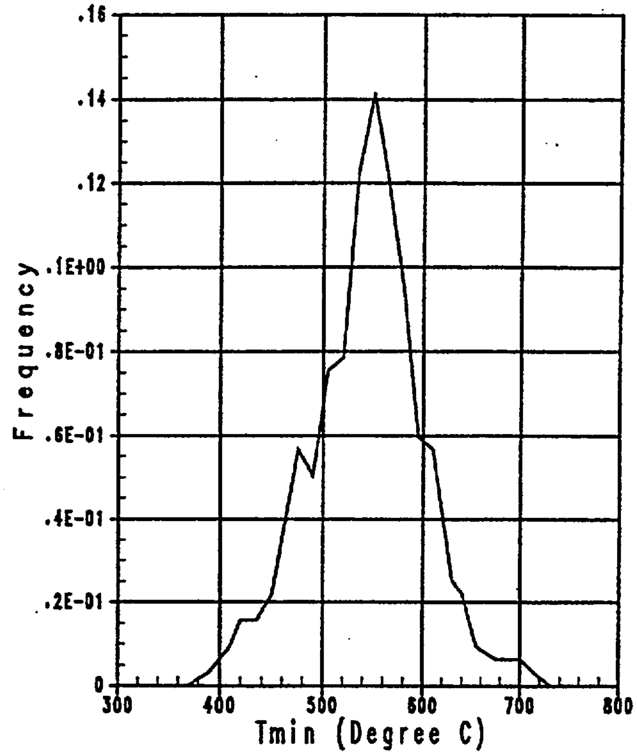

Fig. E-1a. Distribution of experimental rewet temperatures at high mass flow rates.

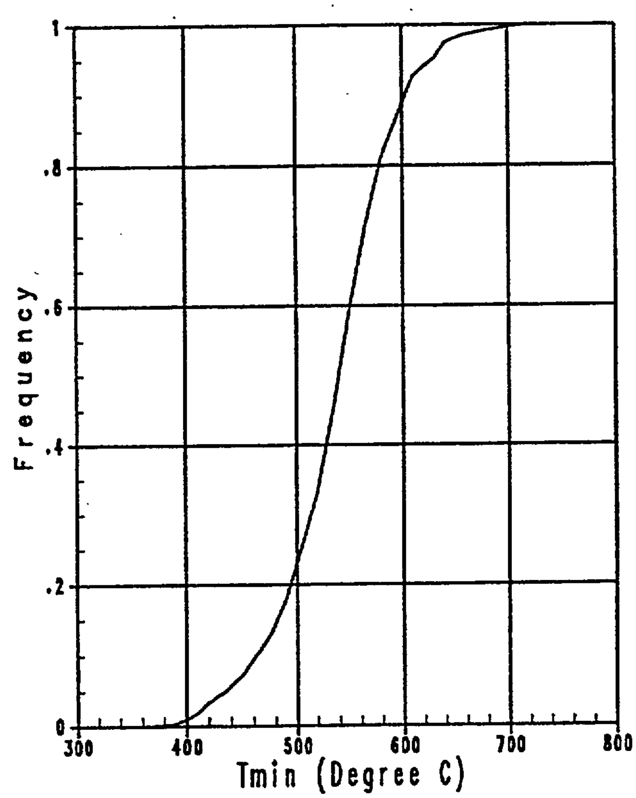

Fig. E-1b. Cumulative distribution for Tmin at high mass flow rates.

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E-2.4. Dryout Effects on High Burnup Fuel (OECD Halden Reactor Project-Norway)

The information regarding this test series was provided by panel-member W. Wiesenack.

Background. The objective of the dry-out test series was to provide information on the consequences for fuel of short-term dry-out incidents in a BWR. The experimental method employed was, on an individual basis, to expose fuel rods with different burnups to single or multiple dry-out events; to follow this by either unloading or continued operation in the reactor; and to finish with post irradiation examination and testing with emphasis on fuel clad properties. The test series was co-sponsored by the Halden Project's joint program and TEPCO (Japan).

Testing program. The test series comprised three loadings of IFA-613. Each rod was contained in a stainless steel channel within the rig so that the coolant conditions for each rod could be controlled individually. In this way separate dry-out scenarios were effected for each rod. Thermocouples attached to the surface of the test rods were used to monitor clad surface temperature and clad elongation was monitored by way of an extensometer. The first and second loading operated for a month after dry-out whilst the rods in the last loading were unloaded directly after the dry-out procedure. In neither case did any fuel failures develop.

The in-pile dry-out experiments with the third (and last) set of fuel rods in IFA-613 were completed in January '98 (HWR-552, HP-1036) and the post irradiation examination (PIE) on all eight rods in the three test series were finished in September '98 (Kjeller hot cell).

Summary of results. In total, 2 rods with fresh Zr-2 and Zr-4 and 6 rods with clad pre-irradiated to 22-40 MWd/kg (Zr-2, Zr-2 with liner and Zr-4) were individually exposed to reduced or no-flow conditions in a heated light water loop within the Halden reactor. Dry-out occurred over the upper region of each rod, with 6 rods developing peak clad temperatures in the range 950-1200°C occurred in the other two rods.

An overview of the condition of the rods in terms of clad surface condition, rod dimensions and hydriding was achieved using non-destructive PIE techniques such as profilometry and neutron radiography. Clad and fuel microstructure and clad mechanical properties were investigated with destructive PIE techniques including ceramography, metallography, microhardness and ring tensile testing. It was observed that whilst dry-out had not affected the fuel microstructure, significant changes had been induced in the clad. These included high temperature corrosion resulting in moderate growth of the outer surface oxide layer and H_2 pick-up (hydriding formation). Some of the rods also exhibited uniform and localised clad creep-down into pellet-pellet interfaces and in the most severely tested rods that clad had undergone the α to β phase transformation. This material exhibited reduced UTS and brittle fracture. However, significant improvements of ductility were observed in clad that had been exposed to less severe in-pile transients where a small α-phase grain structure was retained and hydrogen pick-up was minimal. None of the rods failed, neither during the dry-out phase or the following steady-state normal operation.

Applications. The data obtained will be used to assess and modify existing rules/regulations in member countries on the continued operation with fuel elements subjected to short-term dry-out transients in boiling water reactors.

APPENDIX F

MEMBERS OF THE HIGH BURNUP FUEL PIRT PANEL

Carl A. Alexander

Carl Alexander is Chief Scientist of Battelle's government sectors operation. He has a B.S. in Mathematics from Ohio University, a M.S. in Physics from the same institution, and a Ph.D. in Ceramic Engineering received in 1961 from The Ohio State University. From 1962 to 1985 he was a member of the engineering and graduate faculty of The Ohio State University, with joint appointments as Adjunct Professor of Nuclear Engineering as well as Ceramics and Materials Engineering. He has also served as Adjunct Professor at the University of Maryland and Southampton University in the U.K. His specialty is nuclear fuels and thermodynamics. He performed some of the first loss-of-coolant simulations in the late 1950s early 1960s. He contributed to Wash-1400 in which he showed the importance of cesium iodide as a transport medium in a LOCA. He performed several studies of fission product release with real fuels at very high temperatures and has evaluated a number of complexes involving urania and Zircalloy at very high temperatures.

Jens G. M. Andersen

Jens G. Munthe Andersen is a principal engineer at Global Nuclear Fuel. He has a M.S. in Nuclear Engineering for the Technical University in Denmark and obtained a Ph.D. in Nuclear Engineering from the same institution in 1974. From 1971 to 1978 he was employed by Risø National Laboratory in Denmark. From 1978 Dr. Andersen has been employed by General Electric Nuclear Energy and since January 2000 by Global Nuclear Fuel (a joint venture of GE, Toshiba and Hitachi). He is currently leader of the Methods and Process Development team at Global Nuclear Fuel. Dr. Andersen has 29 years experience in the nuclear field. He has been primarily engaged in developing computer programs for boiling water reactor transient and safety analysis. He has participated in numerous PIRT panels and the application of the CSAU methodology to BWR.

Brent E. Boyack

Brent E. Boyack is the facilitator for the High Burnup Fuel PIRT Panel. He is a registered professional engineer. He obtained his B. S. and M. S. in Mechanical Engineering from Brigham Young University. He obtained his Ph.D. in Mechanical Engineering from Arizona State University in 1969. Dr. Boyack has been on the staff of the Los Alamos National Laboratory for 19 years; he is currently the leader of the software development team, continuing the development, validation, and application of the Transient Reactor Analysis Code (TRAC). Dr. Boyack has over 30 years experience in the nuclear field. He has been extensively engaged in accident analysis efforts, including design basis and severe accident analyses of light water, gas-cooled, and heavy-water reactors; reactor safety code assessments and applications; safety assessments; preparation of safety analysis reports; and independent safety reviews. He chaired the MELCOR and CONTAIN independent peer reviews and was a member of the Code Scaling, Applicability and Uncertainty or CSAU technical program group. He has participated in numerous PIRT panels. He has over 70 journal and conference publications and is an active member of the American Nuclear Society.

Bert M. Dunn

Bert M. Dunn obtained his B. S. in Physics from Washington State University in 1968 and his M. S. in Physics from Lynchburg College in 1973. Mr. Dunn has worked in LOCA and Safety Analysis for the Babcock and Wilcox Company (B&W) and Framatome Technologies (FTI) for 28 years. Mr. Dunn has served as the lead technically for the development of the B&W and FTI LOCA evaluation models for once through and recirculating steam generator plants. He has worked with both deterministic and best estimate LOCA evaluation techniques. He has also been technical lead for method development and application of boron dilution accident methods and pressurized thermal shock evaluation methods. He is currently employed as an Advisory Engineer with responsibility for the development of LOCA and Safety Analysis techniques for evaluation of advanced cladding materials. This includes test specification development, review and correlation of results, and the incorporation of results into requisite analytical methods. Mr. Dunn has been primary author on several company topical reports covering both methods development and accident analysis.

Toyoshi Fuketa

Toyoshi Fuketa is a Principal Engineer in the Fuel Safety Research Laboratory at the Japan Atomic Energy Research Institute (JAERI). He obtained his B. S., M. S. and Ph.D. in Mechanical Engineering Science from Tokyo Institute of Technology, Japan, in 1982, 1984 and 1987, respectively. Dr. Fuketa has been involved in the Nuclear Safety Research Reactor (NSRR) project to study behavior of LWR and research reactor fuels under reactivity accident and severe accident conditions and to evaluate the thresholds, modes, and consequences of fuel failure in terms of the fuel enthalpy, fuel burnup, coolant conditions, and fuel design. His research interests include fuel-coolant interactions, fuel failure mechanisms and transient fission gas behavior. He was engaged in small-scale steam explosion experiments at Sandia National Laboratories, Albuquerque, from 1988 to 1990, as a visiting scientist.

Larry E. Hochreiter

L.E. (Larry) Hochreiter is a professor of Nuclear and Mechanical Engineering at the Pennsylvania State University and does research and teaching in the areas of two-phase flow and heat transfer, reactor thermal-hydraulics, fuel rod design, and nuclear reactor safety. He received a BS degree in Mechanical Engineering from the University of Buffalo and a MS and Ph.D degrees in Nuclear Engineering from Purdue University. While at Pennsylvania State University, Dr. Hochreiter has developed a detailed reflood heat transfer PIRT to guide the design and instrumentation of the NRC Rod Bundle Heat Transfer program, located at Penn State. Before joining the Penn State University in 1997, Dr. Hochreiter was a Consulting Engineer at the Westinghouse Electric Corporation for nearly 26 years and was responsible for the development, testing validation, and licensing of Westinghouse safety analysis methods. He developed the large-break Loss Of Coolant Accident (LOCA) PIRT for the Westinghouse Best-Estimate Methodology. He also participated in and helped develop the Westinghouse small-break LOCA PIRT. Dr. Hochreiter also developed several PIRTs for the Westinghouse advanced AP600 design for the accident analysis methods and presented these PIRTs to the NRC and the ACRS.

Robert O. Montgomery

Robert O. Montgomery received a B.S. Degree in Nuclear Engineering from Texas A&M University in 1984 and a M.S. Degree in Nuclear Engineering from Texas A&M University in 1987. Under sponsorship from Texas A&M University, Mr. Montgomery worked at Belgonucleaire between September 1986 and December 1986 as a research assistant in the area of steady state fuel performance modeling. Mr. Montgomery is currently Manager of the Nuclear Technology Group at ANATECH Corp and is responsible for the modeling and analysis of Light Water Reactor fuel rod and fuel assembly performance under steady state and transient conditions. Since coming to ANATECH in 1988, Mr. Montgomery has been actively involved in many aspects of nuclear fuel performance, including root cause evaluations of fuel rod failures under power ramp conditions, secondary degradation of Boiling Water Reactor fuel following primary failure, and high burnup fuel behavior during Reactivity Initiated Accidents. As Project Manager for the FREY transient fuel behavior program, Mr. Montgomery has been responsible for the advanced modeling and analysis techniques used in FREY for LWR accident events such as LOCA, BWR ATWS power oscillations, RIA, and Power-Coolant Mismatch Accidents. Between 1991 and 1994, Mr. Montgomery was the project leader in the development of analytical methods for the modeling and analysis of the complex thermal, mechanical, and chemical interactions within a BWR fuel rod experiencing secondary degradation. These activities included integrating experimental results from laboratory tests, reviewing and assessing off-gas activity levels from operating experience, and interpreting post-irradiation examination results. Mr. Montgomery also served on the EPRI/NEI Task Force on Reactivity Initiated Accidents between 1994 and 1996 and was the primary technical consultant to this task force on the analysis and interpretation of RIA experiments performed throughout the world. Mr. Montgomery has published numerous journal articles and technical reports related to nuclear fuel behavior modeling techniques and analysis results.

Frederick J. Moody

Frederick J. Moody is a Consulting Engineer in Thermal-Hydraulics, who has participated in numerous NRC - sponsored peer review groups and Technical Program Groups, involving the analysis of postulated nuclear reactor accidents. He received his Ph.D. in Mechanical Engineering from Stanford University in 1971. He completed 41 years of reactor and containment safety analyses at the General Electric Nuclear Energy Division, where he developed various industry-standard analytical models for studies involving pipe and component rupture blowdown of high pressure steam and water mixtures, containment pressure and jet impingement loads, waterhammer forces associated with pipe flow accelerations, dynamic and thermal response of nuclear reactor core components during accident conditions, and fluid-structure interaction of submerged structures. He has taught numerous engineering courses as an adjunct professor for 28 years at San Jose State University, as an in-plant instructor at General Electric, and more recently as an instructor for professional development courses sponsored by the American Society of Mechanical Engineers. He has authored numerous journal papers, written an engineering textbook, *Introduction to Unsteady Thermo-Fluid Mechanics* (Wiley Interscience, 1990), and co-authored *The Thermal-Hydraulics of a Boiling Water Nuclear Reactor*, 2nd Ed., ANS Press, 1993.

Arthur T. Motta

Arthur T. Motta has worked in the area of radiation damage to materials with specific emphasis in Zr alloys for the last fifteen years. He received a B.Sc. in Mechanical Engineering and an M.Sc. in Nuclear Engineering from the Federal University of Rio de Janeiro, Brazil, and a Ph.D. in Nuclear Engineering from the University of California, Berkeley. He worked as a research associate for the CEA at the Centre for Nuclear Studies in Grenoble, France for two years and as a post-doctoral fellow for AECL at Chalk River Laboratories, Canada, before joining Penn State in 1992. The research programs he developed at Penn State include mechanical behavior of Zr alloys, advanced techniques for characterization of Zr alloys, and its oxides, defects in intermetallic compounds and phase transformation under irradiation. He has expertise in transmission electron microscopy, charged particle irradiation, mechanical testing, positron annihilation spectroscopy and theoretical expertise on phase transformations under irradiation and microstructural evolution under irradiation. He has recently authored review articles on amorphization under irradiation and on zirconium alloys in the nuclear industry. He was recently guest editor for a special issue of the Journal of Nuclear Materials, and was a member of a DOE panel to evaluate research needs on radiation effects on ceramics for radioactive waste disposal.

Kenneth L. Peddicord

Kenneth L. Peddicord is Associate Vice Chancellor and Professor of Nuclear Engineering at Texas A&M University. He received his B.S. degree in Mechanical

Engineering from the University of Notre Dame in 1965. He obtained his M.S. degree in 1967 and his Ph.D. degree in 1972, both in Nuclear Engineering from the University of Illinois at Urbana-Champaign. From 1972 to 1975, Dr. Peddicord was a Research Nuclear Engineer at the Swiss Federal Institute for Reactor Research (now the Paul Scherrer Institute) where he worked in the plutonium fuels program. From 1975 to 1981, Dr. Peddicord was Assistant and Associate Professor in the Department of Nuclear Engineering at Oregon State University. From 1981 to 1982, he was a Visiting Scientist at the EURATOM Joint Research Centre in Ispra, Italy where he was involved in the Super Sara Severe Fuel Failure Programme. In 1983, Dr. Peddicord joined Texas A&M University as Professor of Nuclear Engineering. He has served as Head of the Department of Nuclear Engineering (1985-88), Associate Dean for Research (1988-91), Interim Dean of Engineering (1991-93), and Director of the Texas Engineering Experiment Station (1991-93). Since 1994, he has been Associate Vice Chancellor of the Texas A&M University System. Dr. Peddicord serves as the representative of the A&M System to the Governing Board of the Amarillo National Resource Center for Plutonium. Dr. Peddicord's research interests are in the performance and modeling of advanced nuclear fuels. Since 1995, he has been a participant in joint DOE-Minatom activities on excess plutonium disposition and nuclear materials safety. Dr. Peddicord has 120 publications in technical journals and conferences. He is a registered professional engineer in the state of Texas and has been a member of the American Nuclear Society since 1975.

Gerald Potts

Mr. Potts of Global Nuclear Fuel received a Bachelor of Science degree in Mechanical Engineering from the University of California, and a Master of Science degree in Mechanical Engineering from Santa Clara University. Mr. Potts has accumulated 28 years experience in the commercial nuclear power industry within the General Electric Nuclear Energy division. Mr. Potts' responsibilities and experience include fuel rod thermal-mechanical design, fuel rod thermal-mechanical performance and licensing basis analytical model development, and fuel integrity assessment under normal steady-state operation, anticipated operational transient, and accident conditions.

Douglas W. Pruitt

Douglas W. Pruitt is a staff engineer with Siemens Power Corporation. He obtained his B. S. E. from the University of Washington and M. S. in Nuclear Engineering from the University of Michigan. Mr. Pruitt has been on the staff of Siemens Power Corporation 21 years; he is currently a development engineer in Safety Analysis Methods. He has been engaged in both BWR and PWR development including core monitoring, stability measurement and analysis and transient analysis.

Joe Rashid

Joe Rashid is a Fellow of the ASME and a registered Nuclear Engineer. His general field of expertise is computational thermo-mechanics, structural mechanics and material constitutive modeling. He acquired his graduate and undergraduate education in mechanics at the University of California Berkeley, receiving the PhD degree in 1965. Having received his education at the birth place of the Finite Element Method in the early sixties, Dr. Rashid was among the pioneering contributors to its development, in particular three-dimensional computations. Dr. Rashid's three and a half decades career in the nuclear industry began with the gas-cooled reactor technology at General Atomics in San Diego, followed by an eight-year career in BWR technology at General Electric in San Jose, and finally at ANATECH Corp. which he founded in 1978. At General Atomics, his work in the mechanics of concrete reactor vessels and nuclear fuel particles led to the development of the smeared-crack model, which was adopted in finite element codes as the basic model for the cracking analysis of brittle materials. At GE, he was responsible for the development of the industry's first two-dimensional fuel rod behavior code for the analysis of the then-emerging pellet-clad interaction (PCI) problem. At ANATECH, Dr. Rashid undertook the development of the transient fuel analysis code FREY for the Electric Power Research Institute (EPRI). In the aftermath of the Three Mile Island accident, EPRI's collaboration with Sandia in reactor containment research, with Dr. Rashid as the principal investigator for EPRI, led to the institutionalization of the leak-before-break concept for reactor containment structures, thereby profoundly affecting risk assessment of loss of coolant accidents. He participated in severe accident work with Sandia and EPRI, which included the development of constitutive models and analysis methods for the creep rupture of pressure vessel lower head under loss of coolant accident. He participated in the expert review process for NUREG-1150, and was nominated by NRC to chair an international expert panel for OECD'sVessel Investigation Project. Dr. Rashid's publications in the various fields of activity in which he had primary contributions exceed 100, which include journal articles, reports and white papers.

Daniel H. Risher

Daniel H. Risher is a participant on the PWR Reactivity Insertion Accident (RIA) PIRT panel, representing Westinghouse Electric Company. He obtained his B.S. in Mechanical Engineering from the University of Notre Dame. He obtained his Ph.D. in Nuclear Engineering from the University of Virginia in 1969. Dr. Risher has over 30 years of nuclear experience with Westinghouse, in the fields of Systems Engineering, Core Engineering, and Nuclear Safety and Transient Analysis. During this period, he has been responsible for the functional design and evaluation of safety-related systems for Westinghouse PWRs, the calculation of the transient response of the reactors to non-LOCA accident conditions, the safety evaluation of advanced plant and fuel cycle designs, and the preparation of safety analysis reports. Currently, he is a Fellow Engineer in the Transient Analysis group with the responsibility for the development and utilization of advanced transient analysis methods at Westinghouse, using three-dimensional core neutronics methods. Specific areas of expertise include the safety evaluation of PWR response to reactivity-related accidents, including the dropped rod, rod withdrawal and rod ejection accidents.

Richard J. Rohrer

Richard J. Rohrer serves as a member of the High Burn-up Fuel Phenomena Identification and Ranking Table (PIRT) Panel. He is a registered professional engineer in the state of Minnesota. He obtained a B.S. in Nuclear Engineering from the University of Illinois, and an M.S. in Nuclear Engineering from the University of Wisconsin in 1983. He also holds an M.S. in Management from Cardinal Stritch College, and a Senior Reactor Operator Certification for the Monticello Nuclear Generating Plant. Mr. Rohrer has over 16 years experience supporting operations of nuclear power reactors, including licensing, reactor engineering, probabilistic safety assessment, core design, accident analysis, and transient analysis. He currently manages projects for the Monticello Nuclear Generating Plant in the Nuclear Analysis and Design group with Nuclear Management Company. Mr. Rohrer is a member of the American Nuclear Society and has published five technical papers on probabilistic safety assessment and Boiling Water Reactor stability. In addition, he is an active participant in the Electric Power Research Institute's Robust Fuel Program.

James S. Tulenko

James S. Tulenko is Chairman of the Nuclear and Radiological Engineering Department and a Professor of Nuclear Engineering at the University of Florida. He received his B.A. with honors in Engineering Physics from Harvard College and his M.A. in Engineering Physics from Harvard University in 1960. After military service in the Corps of Engineering, he obtained a M.S. in Nuclear Engineering from the Massachusetts Institute of Technology in 1963. In 1980 he obtained a M.E.A. from George Washington University. Professor Tulenko's professional activities have included all aspects of the nuclear fuel cycle. He has over 35 years of experience in fuel design, fuel operation and fuel performance. Professor Tulenko was Manager of Nuclear Development at United Nuclear Corporation where he patented the water hole thermalization concept now utilized in all boiling water reactors. He also was project engineer for one of the first Plutonium reloads in a commercial reactor. He served as Manager of Physics for Nuclear Materials and Equipment (NUMEC) Corporation where he headed up nuclear physics activities. He later served as Manager of Physics and Manager of Nuclear Fuel Engineering for the Nuclear Power Division of Babcock and Wilcox. In 1979 he was made a Fellow of the American Nuclear Society (ANS) for his contributions to the fuel cycle. In 1980 he received the Silver Anniversary Exceptional Service Award of the ANS for his outstanding contributions to the Nuclear Fuel Cycle in the first 25 year of the ANS. In 1997 he received the Mishima Award of the ANS given for outstanding contributions to Nuclear Material Research. He also was awarded the Glenn

Murphy Award of the American Society of Engineering Education given to the Outstanding Nuclear Engineering Educator. He is a Board Member of the National Nuclear Accrediting Board of the Institute of Nuclear Power Operations and a Board Member of the American Nuclear Society. He is also a Commissioner of the Engineering Accreditation Commission. He has over 100 journal and conference publications and has consulted for a variety of government agencies and commercial companies.

Keijo Valtonen

Keijo Valtonen is a Chief Inspector with the Radiation and Nuclear Safety Authority of Finland. He obtained his degree from the University of Helsinki where he majored in reactor physics and thermal hydraulics. His primary duties since 1975 have been fuel, nuclear and thermal-hydraulic design of reactor cores; transient and accident analysis for Loviisa (VVER-440 type PWR) and Olkiluoto (ABB-Atom type BWR); and operator qualification, including oral licensing examinations and review of operator instructions. He has reviewed plant feasibility studies, including those for the VVER-1000, ABB-Atom BWR 90, Siemens PWR, and SECURE and PIUS. He has reviewed numerous feasibility studies for new fuel designs, including VVER Zr 1% Nb, BNFL-VVEF fuel, ABB 8x8, SVEA 64, SVEA 100, Siemens 9x9, GE12 and Siemens ATRIUM 10. He has participated in safety reviews for the RBMK. He has engaged in research work on the transient behavior of BWR and PWR reactor cores, BWR stability analysis, validation of TRACB and RAMONA computer codes, PWR boron dilution, and several fuel transient behavior studies for VVER and BWR reactors. He has been engaged in international cooperative efforts including IAEA and OECD development of safety criteria for future nuclear reactors, regulatory approaches to severe accident issues for the OECD/CNRA, a state-of-the-art report on BWR stability, the European Union's safety RBMK safety review, the OECD/CSNI task force of fuel safety criteria.

Wolfgang Wiesenack

Wolfgang Wiesenack is the acting general manager of the OECD Halden Reactor Project. He obtained an MS in nuclear engineering from the University of Hanover, Germany, in 1976 and a PhD in nuclear engineering and LWR fuel behavior modeling from the same university in 1983. Dr. Wiesenack had a research assistant position at the University of Hanover, working on LOCA analysis (RELAP 4) and modeling of LWR fuel behavior in normal operating conditions. He joined the OECD Halden Reactor Project in 1984. As senior reactor physicist he was responsible for the core physics calculations of the Halden reactor, including nuclear design studies of experimental rigs, core loadings and updating of the reactor's safety report. He was also responsible for the data acquisition of the reactor and implemented a completely renewed system. As the head of the Data Acquisition & Evaluation division, he was in direct contact with many aspects of fuels and materials behaviour under steady state and ramping and transient conditions. He was actively engaged in the execution of the IAEA code comparison exercise FUMEX to which the Halden Project provided the data. He was also a member of the FRAPCON peer review team. He is a member of the German nuclear society.

APPENDIX G

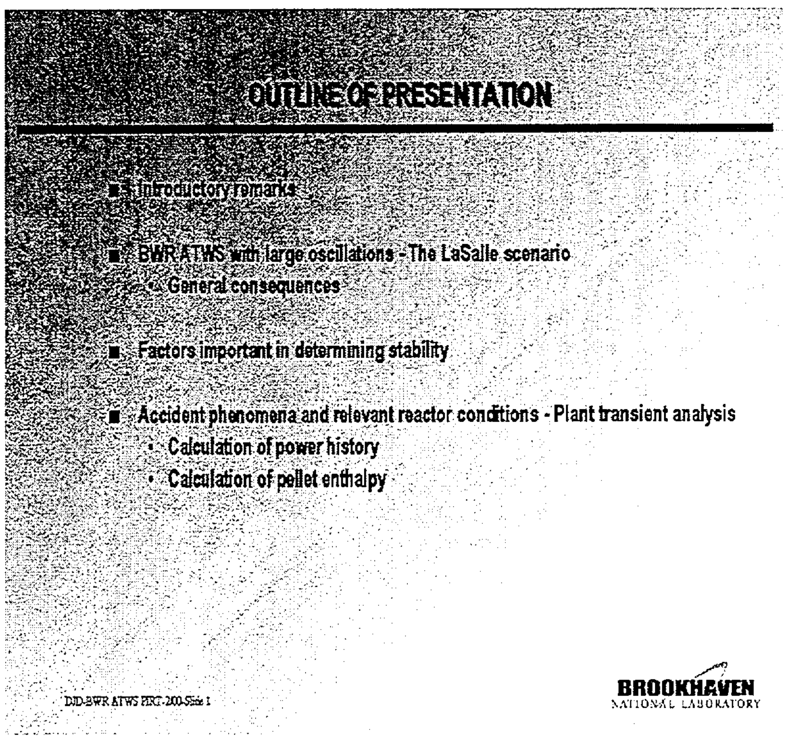
TUTORIAL PRESENTATIONS TO THE PIRT PANEL

This appendix contains information presented to the PIRT panel with the objective of assisting the panel members to develop a common understanding of power oscillations without scram in a boiling water reactor containing high burnup fuel.

G-1. BWR ATWS Events with Large Oscillations

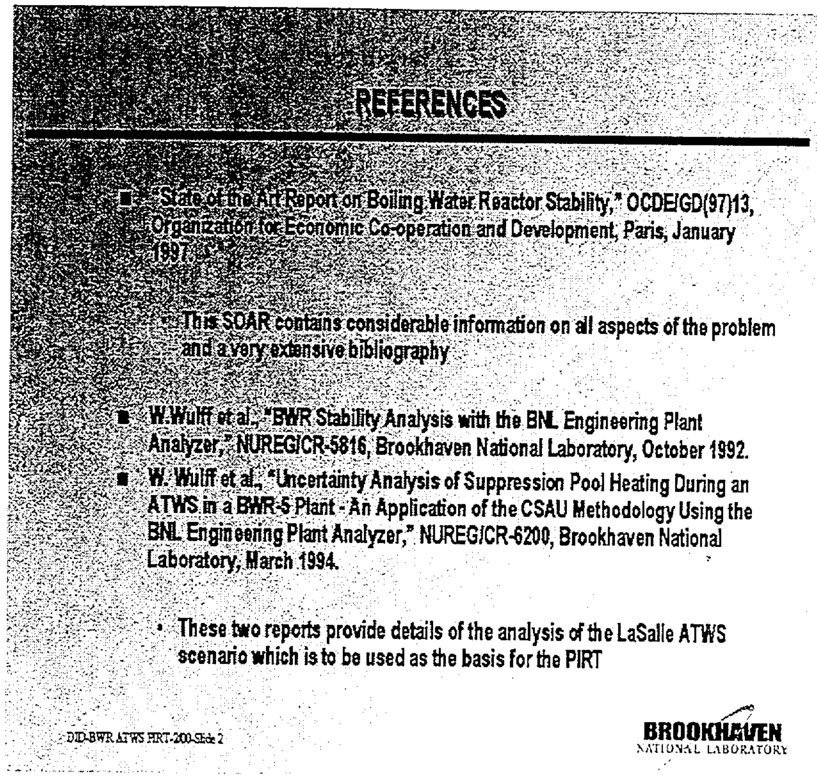
This review was prepared for the PIRT panel by David J. Diamond.

Outline of Presentation (Slide 1)



After some introductory remarks, I will get into the essence of my presentation: to explain what we're calling the LaSalle scenario, which will be the basis for the PIRT exercise. That, of course, is the BWR ATWS with large oscillations. I will go through in some detail the consequences of that event and how that event progresses. I will also spend some time talking about important factors in determining stability and which factors lead to these large oscillations. Finally I will take everything that I've discussed up to that point and put it in the context of the accident phenomena and relevant reactor conditions, which I think are important in the first PIRT category, Plant Transient Analysis. For this category, I will also use the same two elements that we used for the PIRT on reactivity initiated accidents, namely, the calculation of power history and the calculation of pellet enthalpy.

Slide 2



I'd like to first point out a couple of references. The first one is a fairly recent state-of-the-art report (SOAR) on BWR stability. It was done by the OECD NEA in 1997, and contains considerable information on all aspects of the problem of BWR instability. It has a very extensive bibliography.

Keijo Valtonen was one of the authors. Siegfried Langenbuch, who is not here, was another author. I don't know if there are any other authors present today. It's a very useful source of information. Those of you in the field have probably seen it. For those of you not in the field but who want to get more information, I suggest taking a look at this report.

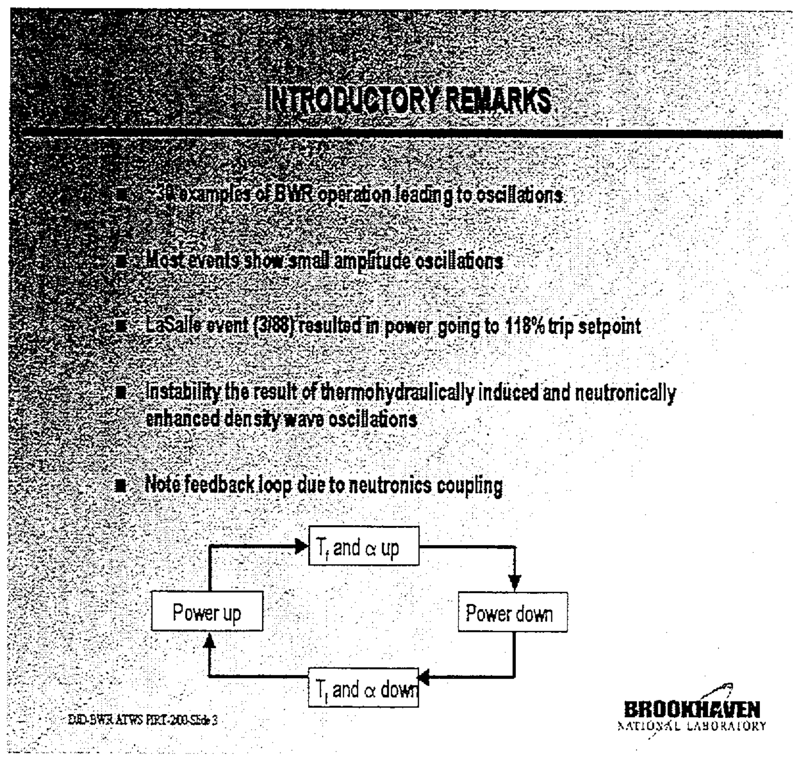
Fuel behavior is addressed in this document, but of all the subjects having to do with BWR instability, it's the one that's treated in least detail. I think we are getting into an area with this PIRT panel in which there hasn't been that much documented, and I think that's why we need the judgment of this panel.

The two other references on this page are Brookhaven National Laboratory (BNL) reports which discuss a set of calculations of different ATWS scenarios.

The one I'm going to be focusing on is based on what happened at the LaSalle plant. The reports are available if you need more details.

I was not involved in these studies, but the graphs from these reports used in my presentation explain the behavior of the core during a BWR ATWS and are based on the calculations done with the Engineering Plant Analyzer at Brookhaven.

Slide 3

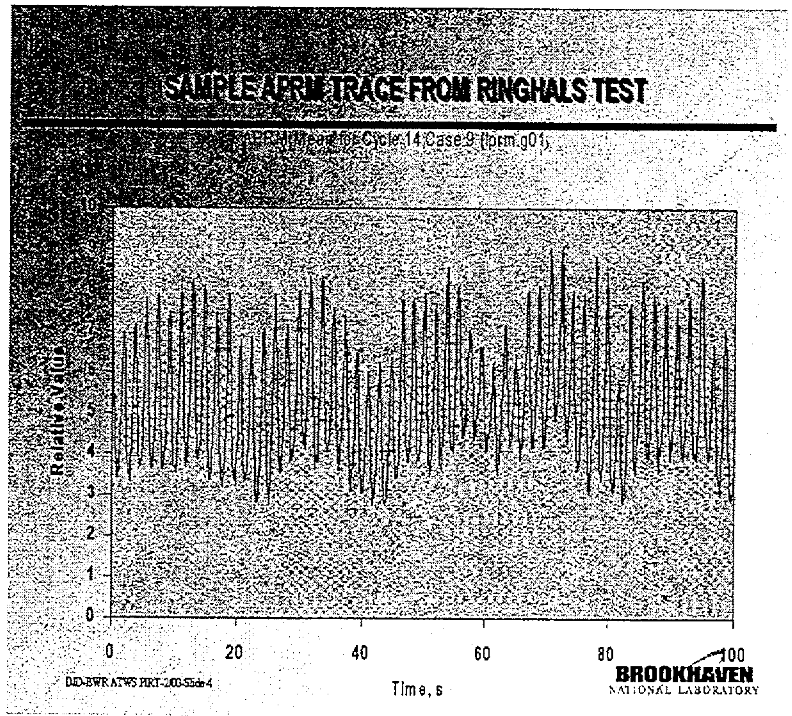


The slide shows that there have been about 30 examples of BWR operation leading to oscillations, but actually there have only been about a dozen examples where plants went into an oscillatory mode unexpectedly. The remaining examples of oscillatory behavior were forced upon the reactor in order to provide test data to understand stability.

Most of these events show small amplitude oscillations;

Slide 4 is an example of a trace. This is the APRM signal from one of the tests done at the Ringhals plant, and it's a typical small amplitude oscillation, where the power varies by about plus or minus 20 percent.

The oscillations look sinusoidal. In this particular case, you can see that they're somewhat modulated, but that's a typical small amplitude oscillation that one would normally expect to see under unstable conditions.
Slide 4



Now, the LaSalle event in 1988 was a little bit different because it actually resulted in the power going fairly high and getting to about 118 percent. I'll explain that scenario in more detail below.

Instabilities are the result of thermal-hydraulically induced and neutronically enhanced density wave oscillations. The feedback loop of most interest here is really quite simple (see Slide 3). If the void fraction goes up, then you expect, because of the negative effect on reactivity, that the power would go down. If the power goes down, then the void fraction goes down; also the fuel temperature goes down. That has a tendency to make the power go up, because now you're increasing reactivity and, of course, if the power goes up then the fuel temperature and the void fraction goes up, and so on and so forth.

Normally, these types of perturbations are damped, and one doesn't have to even think about a repeating feedback cycle. If you're in a region of instability, this type of behavior can continue and one can get a limit cycle where the amplitude is not damped, and oscillations continue.

The LaSalle Scenario

Now, let's start to take a look at a little bit more detail of the LaSalle scenario (Slide 5). The plant was initially at 85 percent of rated power, 75 percent of rated core flow. There was a problem with technicians making an error while they were doing some maintenance, and that error propagated to the point that the recirculation pumps tripped, causing other consequences at the plant and leading to the feedwater heaters being isolated.

Slide 5

"THE LASALLE SCENARIO"
A BWR ATWS WITH LARGE OSCILLATIONS

ACTION OR EVENT	TIME, min
Steady state: 85% of rated power; 75% of rated core inlet flow	0
Recirculation pumps tripped, feedwater heaters isolated	0
Reactor power drops to 40%	0.5
Core flow reduced to natural circulation (29% of rated flow)	0.8
Reactor power increases to 45%; small oscillations start	1.2
Modulated oscillations start	5.5
Power and flow oscillations begin to grow rapidly	6.1
Reactor trip setpoint (118% P) reached but assumed to fail	7.0
Power oscillations reach limit cycle amplitude	12.
End of simulation	15.

DJD-BWR ATWS PIRT-200-Slide 5

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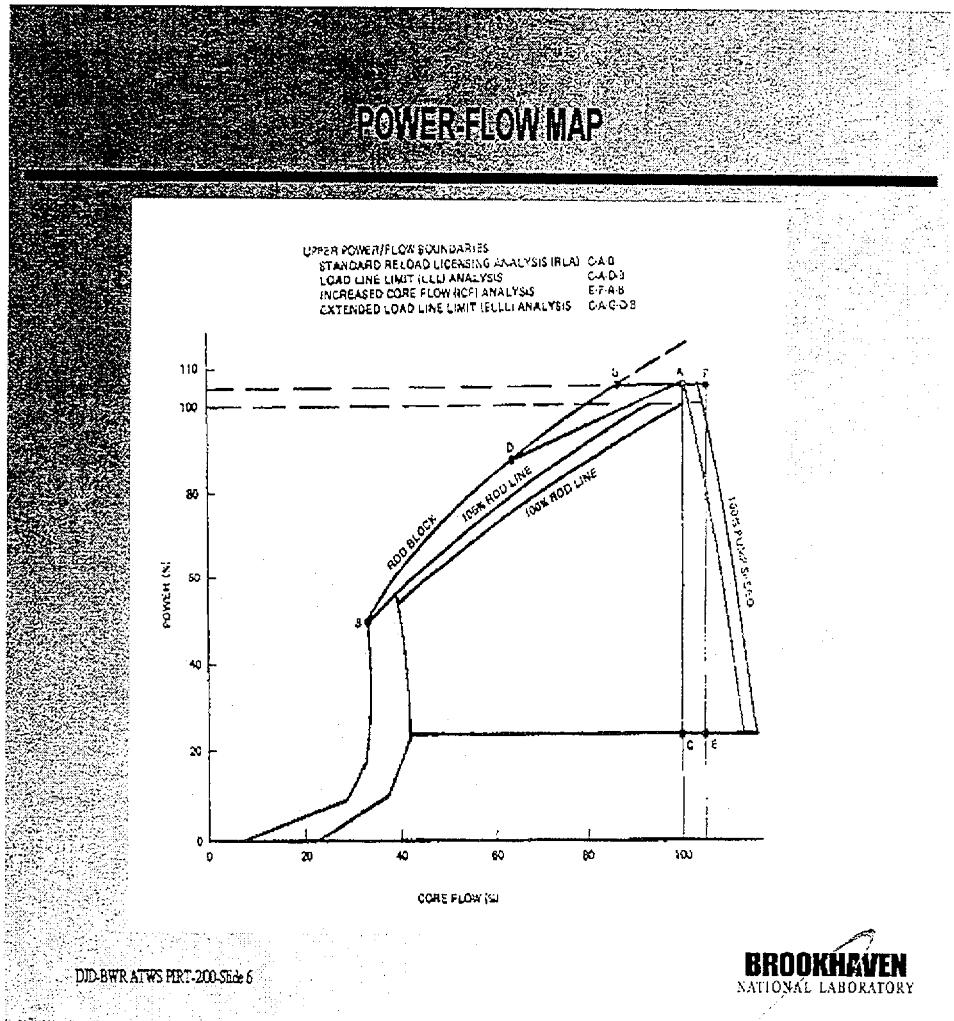
The recirculation pump trip leads to an increase in voids in the core, the reactor power drops to about 40 percent, and the core flow is reduced to natural circulation conditions, about 29 percent of rated flow. This all happens within the first minute.

Actually, though, the reactor power begins to increase a little bit to about 45 percent because the sub-cooling of the feedwater flow into the core increases because the feedwater heaters are isolated, and the feedwater becomes a little bit cooler. Then at about 1.2 minutes, some small oscillations start.

Consider the power-flow map shown in Slide 6.
Slide 6



This map is not from the LaSalle plant, but all of these power-flow maps are very similar. What you had is a situation in which you were at 85/75 (power/flow) initially. You trip the pumps, you come down a trace defined for fixed control rod position, the core flow and power decreases, and you come down towards the natural circulation line.

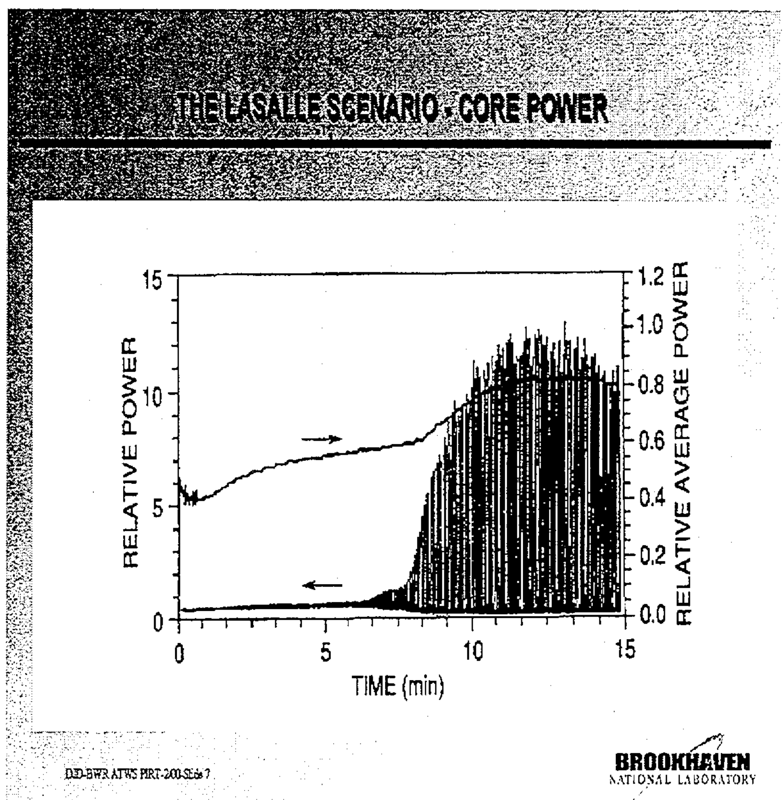
It was stated that the oscillation started when the reactor power increased to about 45 percent and the core flow was about 30 percent. Generally, if you look at a power-flow map like the one shown, you find that the region of instability is in this area (i.e., close to the 45/30 point). It's generally an area of low flow and relatively high power, which is what happened at LaSalle.

At a little over one minute, the oscillations started. They were modulated for a while, and at about six minutes they started to grow rapidly, to the extent

that at seven minutes reactor trip occurred at the plant at 118 percent of nominal power. At LaSalle the operators tried to restart the recirculation pumps, and they were about to manually scram the reactor when the automatic reactor trip occurred.
In our scenario for the PIRT exercise, we're going to assume that trip failed and that the oscillations continued. The calculations I'm going to present show the amplitude of these oscillations continuing to grow. They reached limit cycle amplitude in about 12 minutes, and I will show calculations up to only 15 minutes. We might propose that at 15 minutes, the operators could apply sufficient procedures in order to shut down the reactor. At that point in time, there will have been so many oscillations that if fuel behavior was a concern, you would have already worried about it.

Slide 7 shows the calculated power as a function of time.

Slide 7



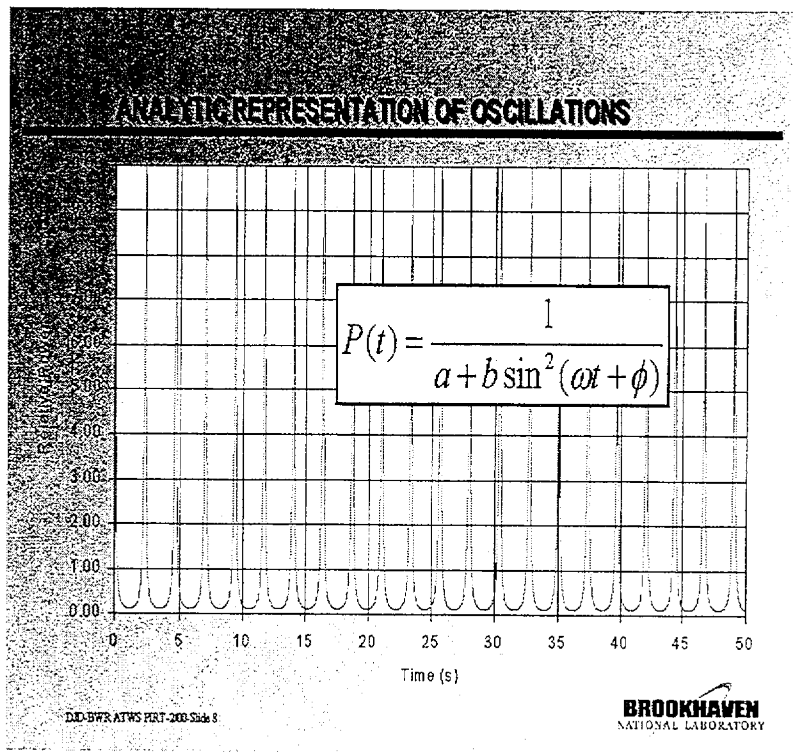
The ordinate on the right side will be discussed later. The nominal total core power on the graph is 1.0. The graph shows how these oscillations start out and then grow to quite large amplitude.

The calculations used to obtain Slide 7 were done at BNL with the Engineering Plant Analyzer, which combines a point neutron kinetics model with the thermal-hydraulic model of the entire nuclear steam supply system. It also models the relevant control systems.

Details of an ATWS Event with Large Oscillations

Slide 7 indicates the general behavior. We can focus a little bit more on these oscillations by looking at an analytic representation (derived at BNL) and looking at a blow-up of only 50 seconds (Slide 8).

Slide 8.

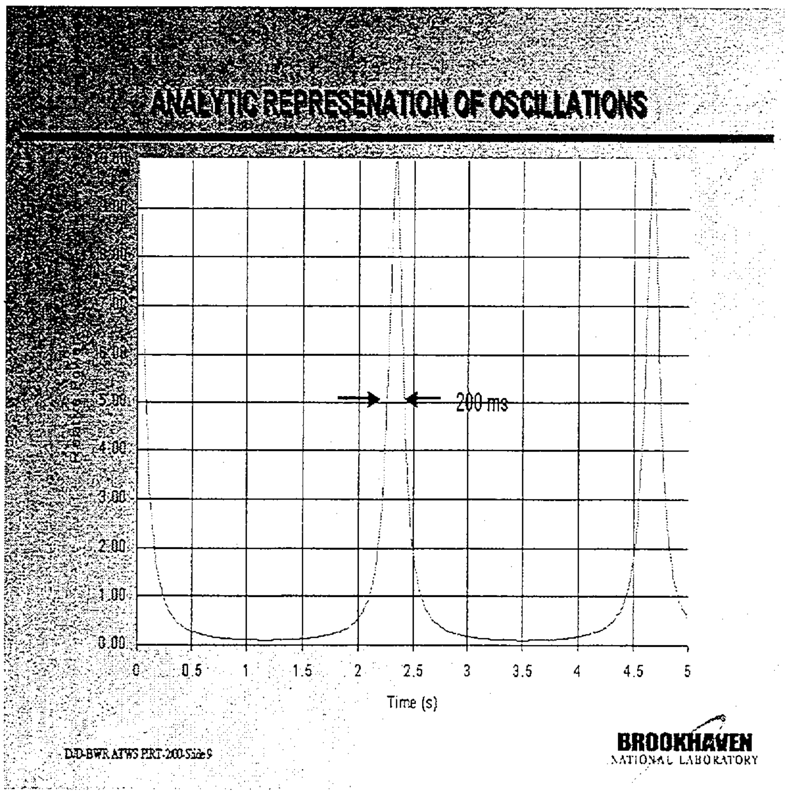


Before, we were looking at 15 minutes. I find it useful to use this function to study some of the properties of these oscillations. This function is based on what we see, not just in the Brookhaven calculations, but in calculations that other people have done to study ATWS.

It's a simple function: the two parameters, a and b, are determined by the maximum amplitude desired for power and the average power desired. In the case shown, the maximum is ten times nominal, and the average power is just above nominal power (1.0).

Slide 9 shows these oscillations on a smaller time scale (0-5 s). The oscillations are sharp power pulses, separated by a rather broad trough. For this case, the full width at half maximum is about 200 ms. When we discussed the rod ejection accident (REA) for a PWR, we were talking about

Slide 9



pulses of about 50 to 75 ms. The pulses on Slide 9 are quite broad relative to the REA situation, but still what I would consider quite sharp in terms of pulse width.

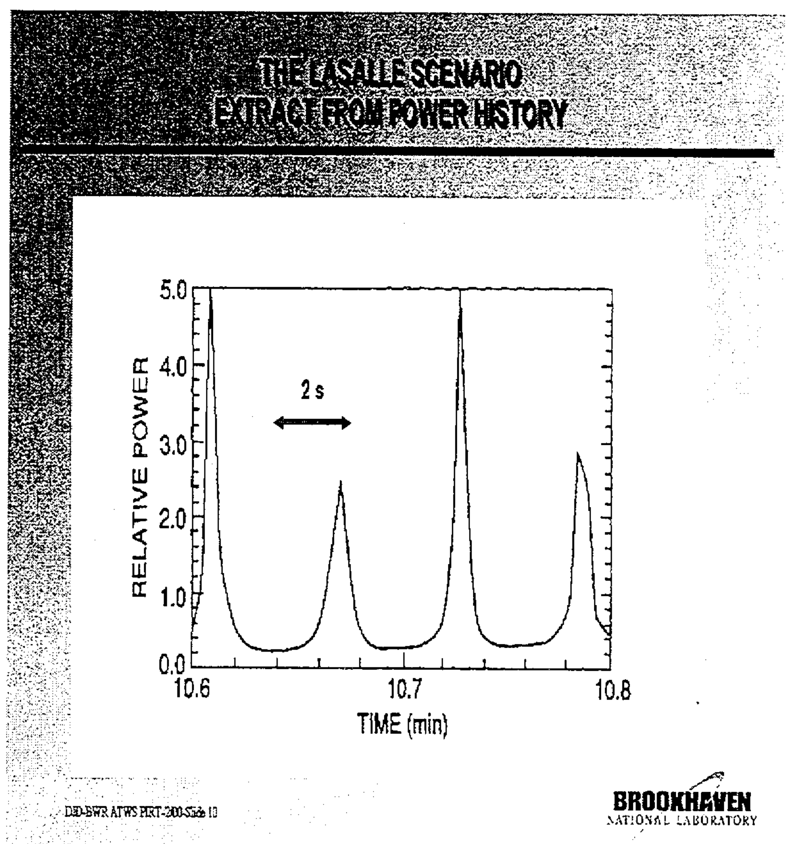
The period of the oscillations is a little bit more than two seconds, and that corresponds to a frequency in the range of 0.4 to 0.5 Hz. Of those BWR oscillations that have been observed in plants, all of them tend to be in that frequency range, and this is determined essentially by the thermal-hydraulics of the BWR.

The oscillations observed at LaSalle and present in the ATWS simulation that is the basis of the PIRT exercise are global oscillations--as most of the BWR instability situations have been--where the entire core is in phase. There have been regional oscillations observed in some plants where parts of the core are out of phase with other parts. For the situation with core-wide oscillations you can get a reasonable representation of the oscillation with point kinetics models. In general, however, three-dimensional neutron kinetics models are needed for a good representation of core behavior.

Continuing with the discussion of the behavior during the event, recall that the power oscillates between 30 and 1300 percent in this particular calculation. Note, too, that a bifurcation pattern is seen during the increase in power peaks, which might be of interest in terms of fuel behavior. During a ten minute time period, what was observed in the calculations was a bifurcation pattern which shows up as one large pulse, followed by a short pulse, followed by a large pulse, followed by a short pulse, with that pattern persisting for a period of time.

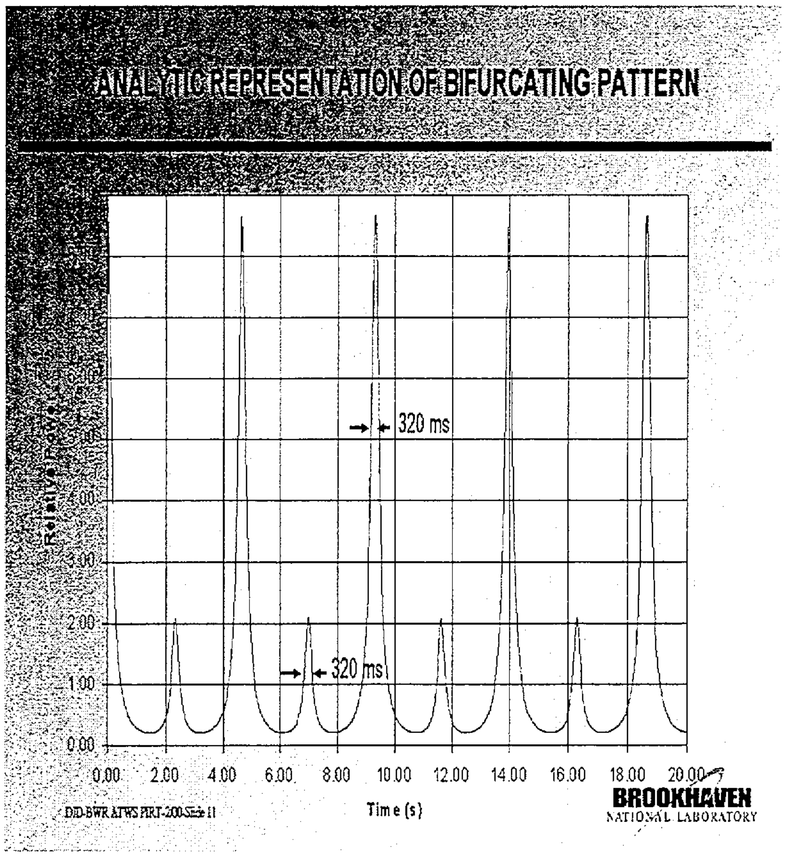
Slide 10 shows this pattern. This graph may be somewhat deceptive because the peak power actually went up to about ten at about ten minutes. Unfortunately there weren't sufficient calculational points plotted here, so the power is only seen going up to a relative power of five. The bifurcation pattern has been observed by many analysts looking at large oscillations.

Slide 10



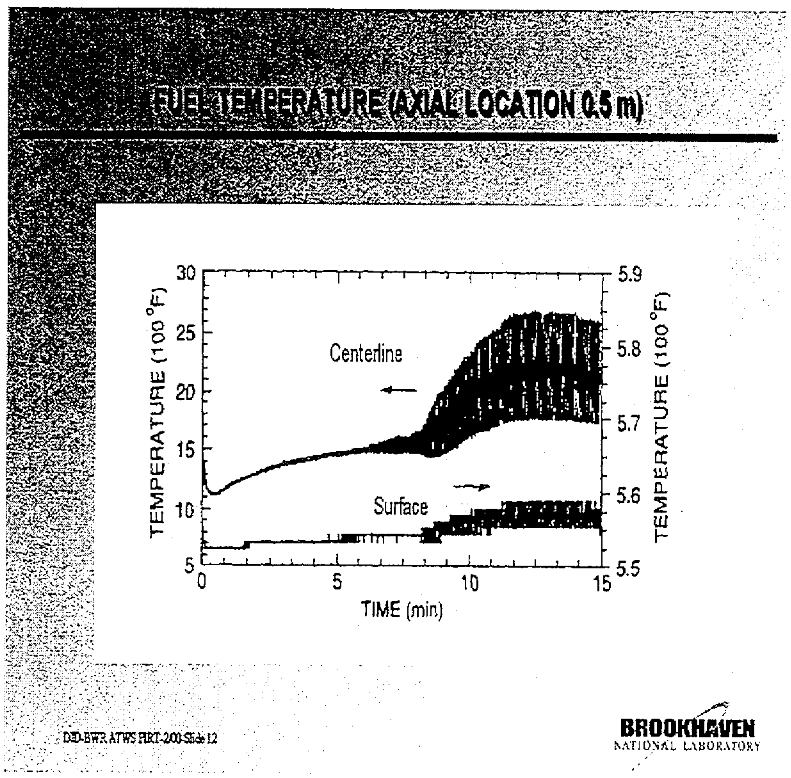
If one looks at the corresponding analytic representation on Slide 11, one sees pulses with full width at half mast that are a bit more than 200 ms, in this case, 320 ms. Hence, in the LaSalle scenario, the range of pulse widths could be as large as about 100 ms.

Slide 11



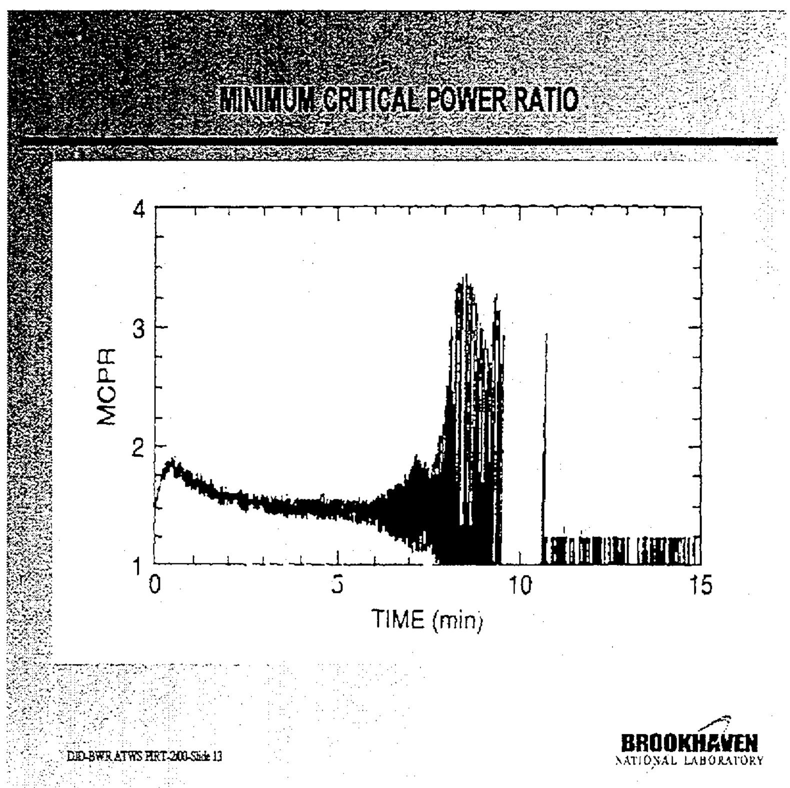
Slide 12 looks at fuel temperatures corresponding to the power in Slide 7. Temperature is of particular interest in determining fuel behavior. Fuel centerline temperature varied between 927 °C and 1480 °C, and the fuel cladding temperature only varied by about six degrees C. These are core average temperatures; local temperatures will be different. Note that the model does take into account a radial distribution in averaging parameters, and the thermal-hydraulic calculation is done with an average axial distribution of power.

Slide 12



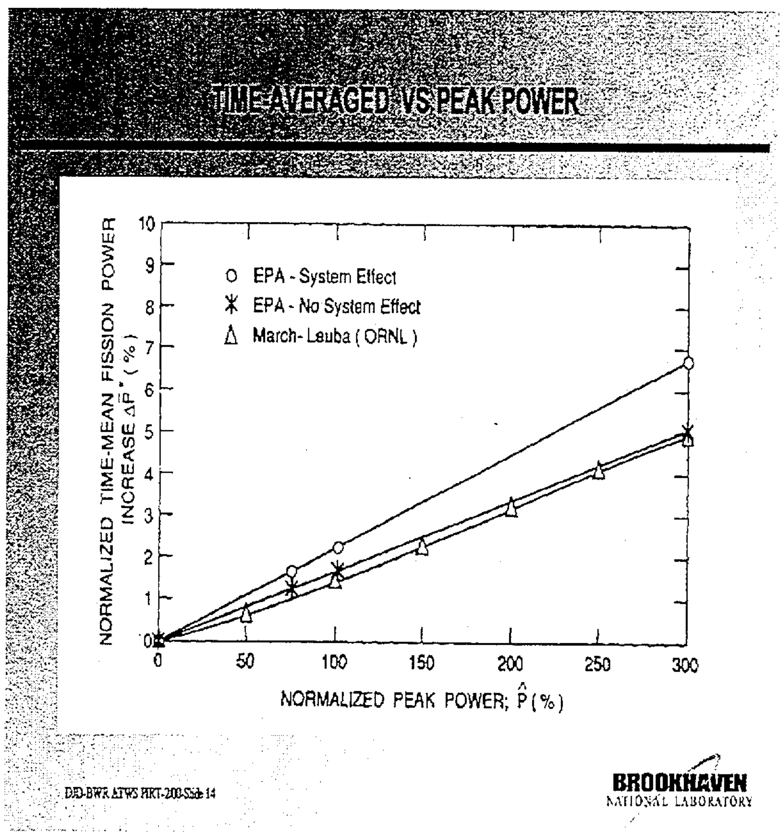
The critical power ratio (CPR) limit was reached temporarily in this calculation, and Slide 13 is the minimum CPR, in the hot assembly. When that's below 1.05 or certainly below 1.0, it suggests a critical heat flux situation somewhere in the core. This is one of the areas where the modeling is problematic because you're talking about using a critical heat flux ratio in a situation with oscillatory heat flux. Although the calculation is problematic, it suggests that it is possible to get to critical heat flux. Note that the CPR is calculated for the hot channel and hence does not feed back into the reactor calculation, which is based on the average channel. The post critical heat flux models, including rewetting, have large uncertainties but also do not feed back into the analysis.

Slide 13



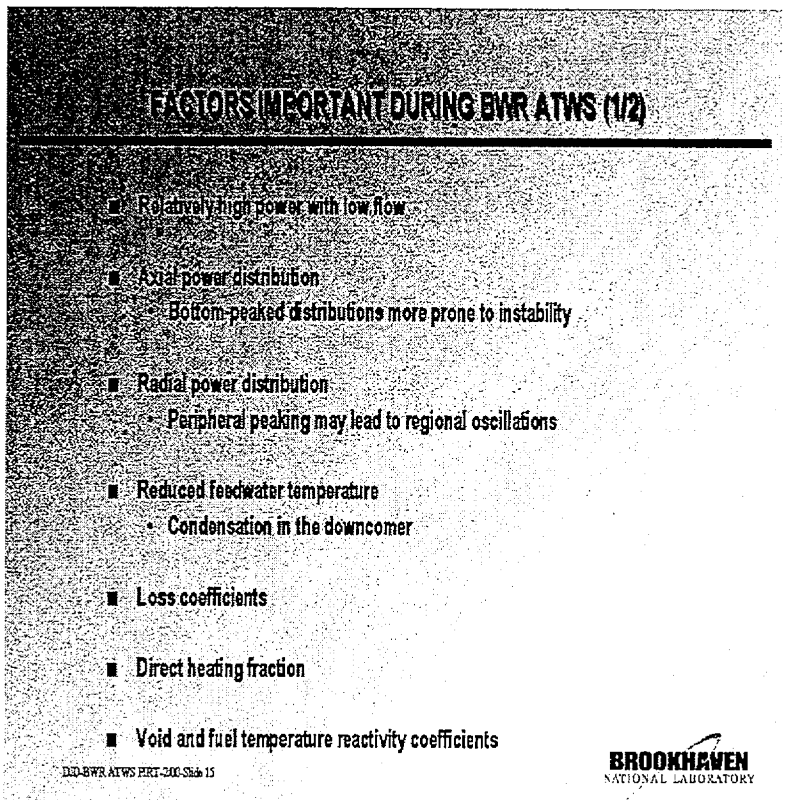
During the scenario, the time-averaged power rises about two percent for every 100 percent increase in peak power (see Slide 14). Time averaged power is the power averaged over a 60 second interval. Slide 7 shows the average core power, and as shown at the beginning of the event, it has dropped down to about 40 percent of nominal power. At the end of the event, the average power is about 80 percent of nominal power. Despite the fact that one has these very large amplitude oscillations, the average power in the reactor is actually less than the nominal power.

Slide 14



Some factors important during this BWR ATWS (see Slide 15) were discussed in the original BNL report, but they have also been discussed by many other people in the literature. I'll go through them very briefly. We already discussed the relatively high power and low flow condition, which brings on the onset of the oscillations. What hasn't been mentioned is the axial power distribution; it's been found that bottom peak distributions are more prone to instability. The LaSalle plant indeed had a bottom-peaked power distribution at the initiation of the oscillation.

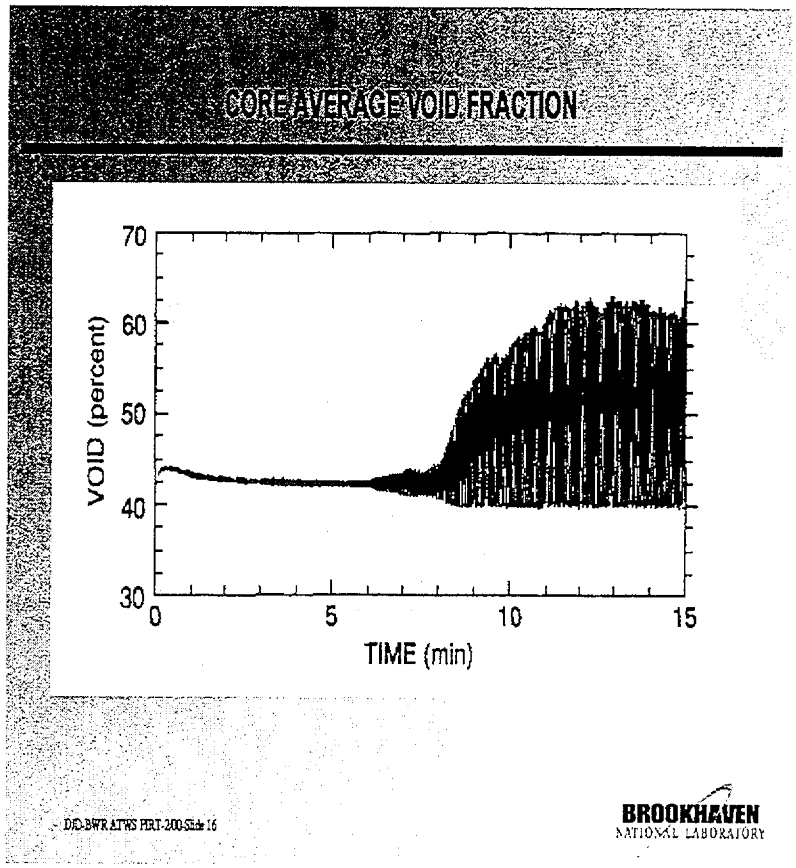
Slide 15



I feel the effect of radial power distribution has not been studied as well as the effect of axial power distribution. What is seen is that peripheral peaking may lead to regional oscillations. The reduced feedwater temperature that occurred at LaSalle was important in increasing the power, and as mentioned, anything that increases the power and reduces the flow at the same time tends to lead to a destabilizing mode. If instead of reducing feedwater temperature, a control rod were withdrawn, it might have the equivalent effect. Note that in modeling feedwater temperature, one of the areas most difficult to model which determines the temperature at the core inlet, is the condensation in the downcomer.

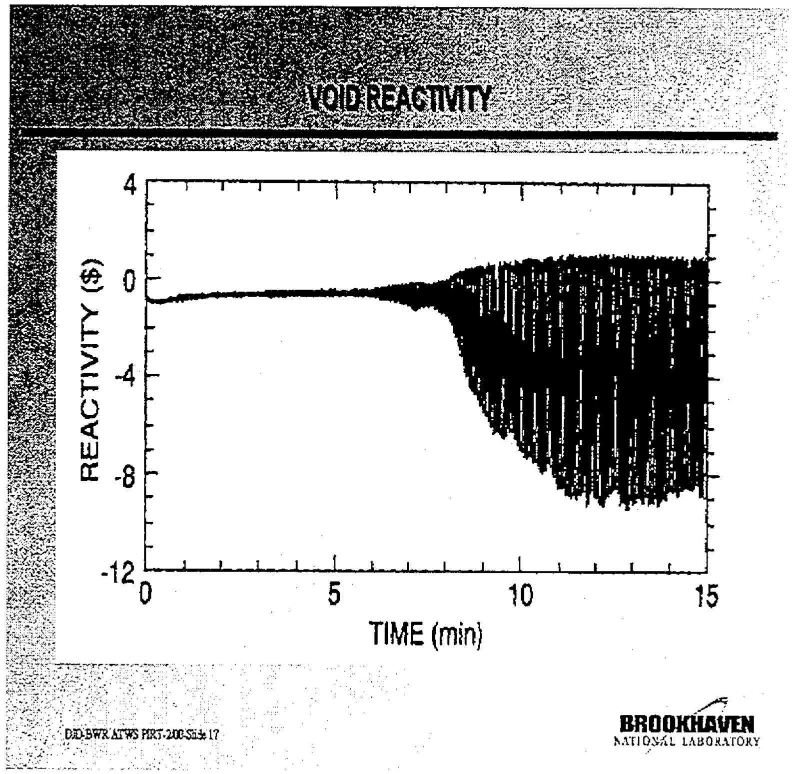
The loss coefficients in the core are important because they determine the pressure drop, which, of course, impacts the flow distribution into the reactor. Direct heating fraction always has a bearing. Void and fuel temperature reactivity coefficients are important. Slide 16 shows the core average void fraction. The nominal void fraction during power operation is about 40 percent. The void fraction varied considerably during this event, and with the average maximum shown here as above 60%, one can speculate that perhaps void fractions indicating dryout are obtained in some parts of the fuel assembly.

Slide 16



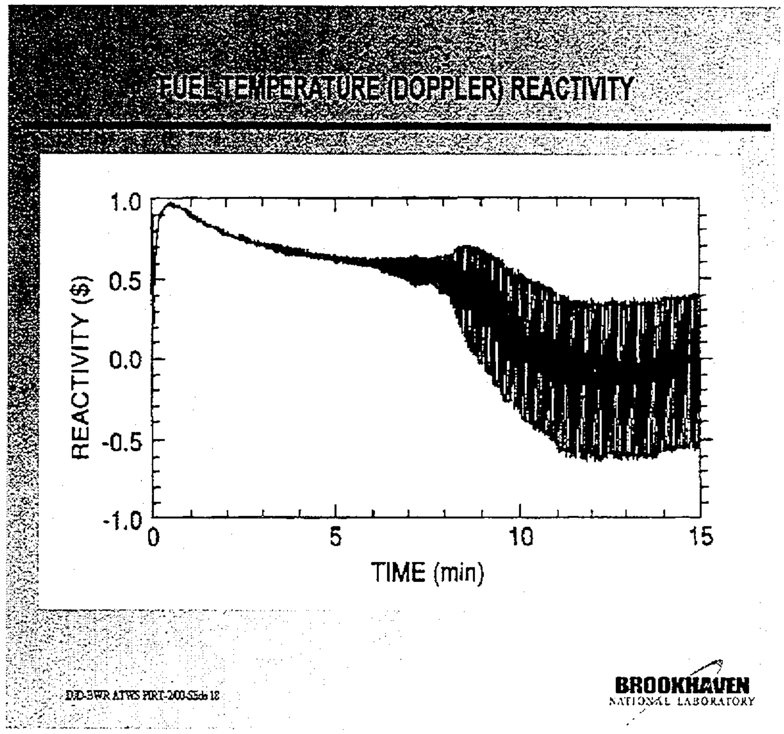
The variation in void fraction leads to a large change in void reactivity (Slide 17). The average value is around minus four dollars.

Slide 17



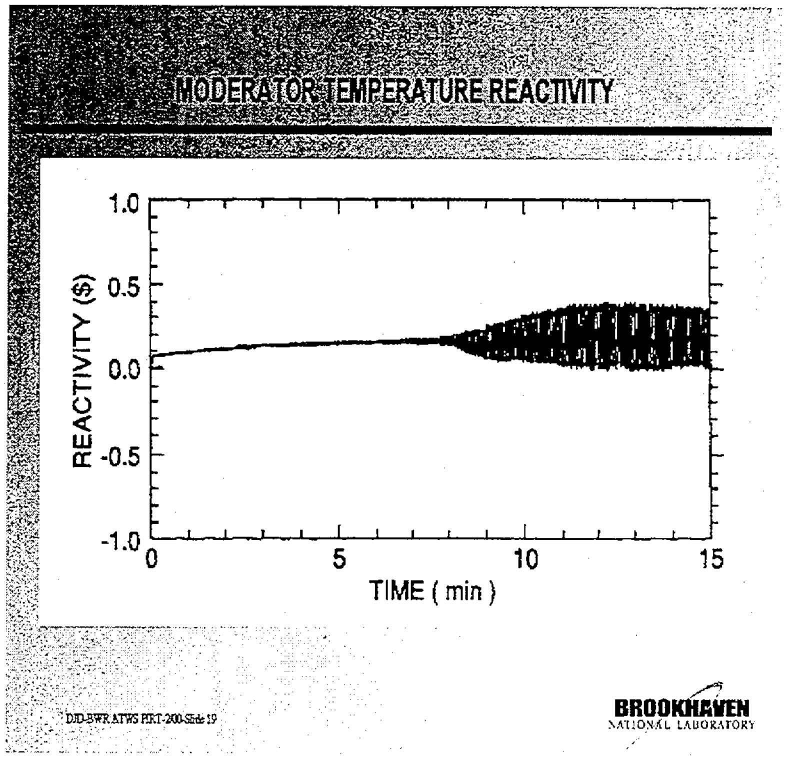
The fuel temperature, or Doppler, reactivity is shown in Slide 18. It varies by about plus or minus a half dollar.

Slide 18



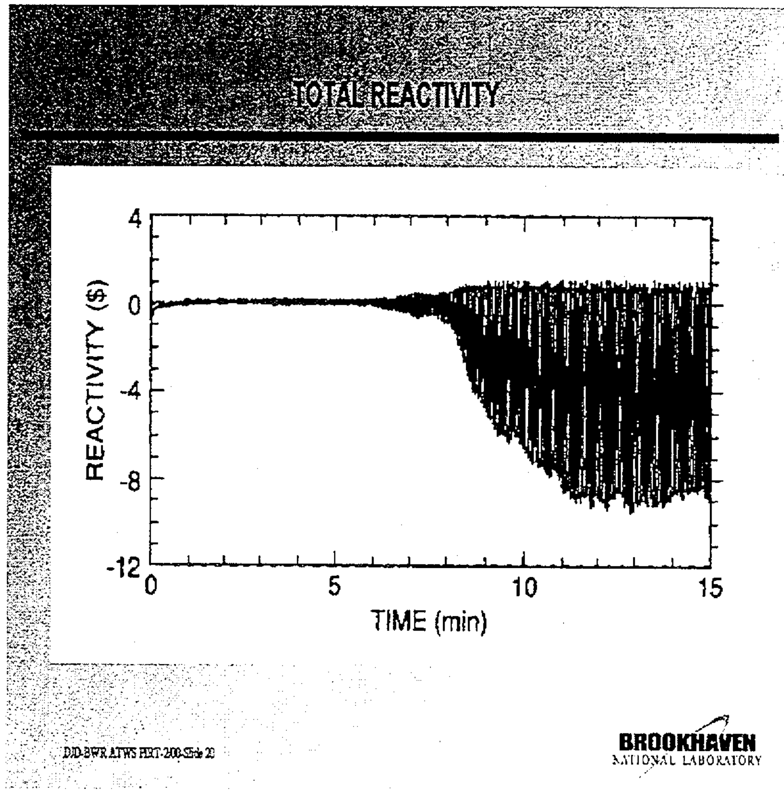
The moderator temperature reactivity, which is the result of heat flux into the liquid as well as variations in the pressure which change the saturation temperature, is shown in Slide 19 and is not large relative to the other components of reactivity.

Slide 19



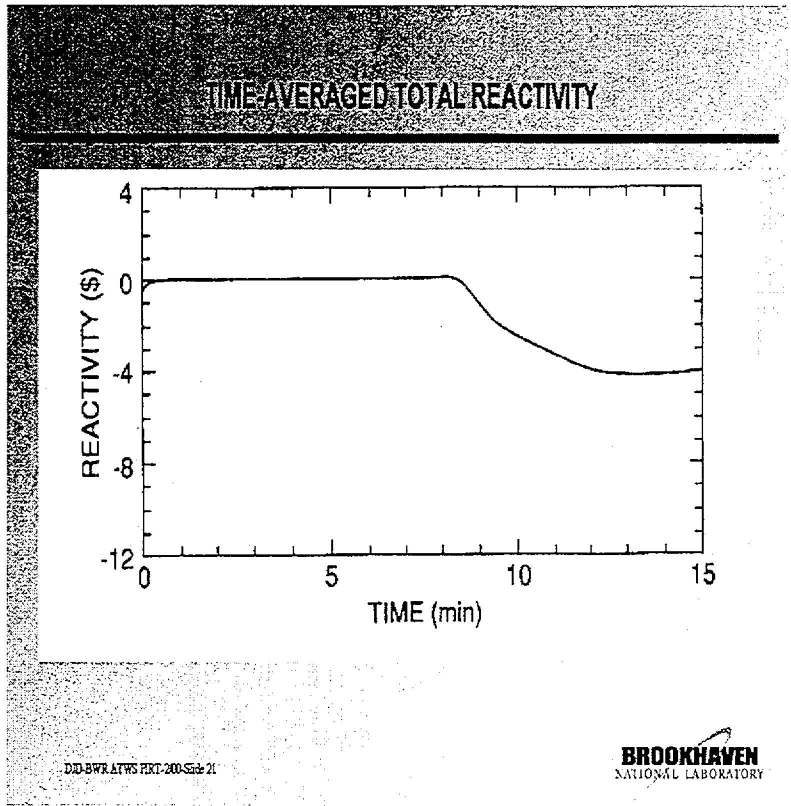
The net reactivity or the total reactivity is given in Slide 20.

Slide 20



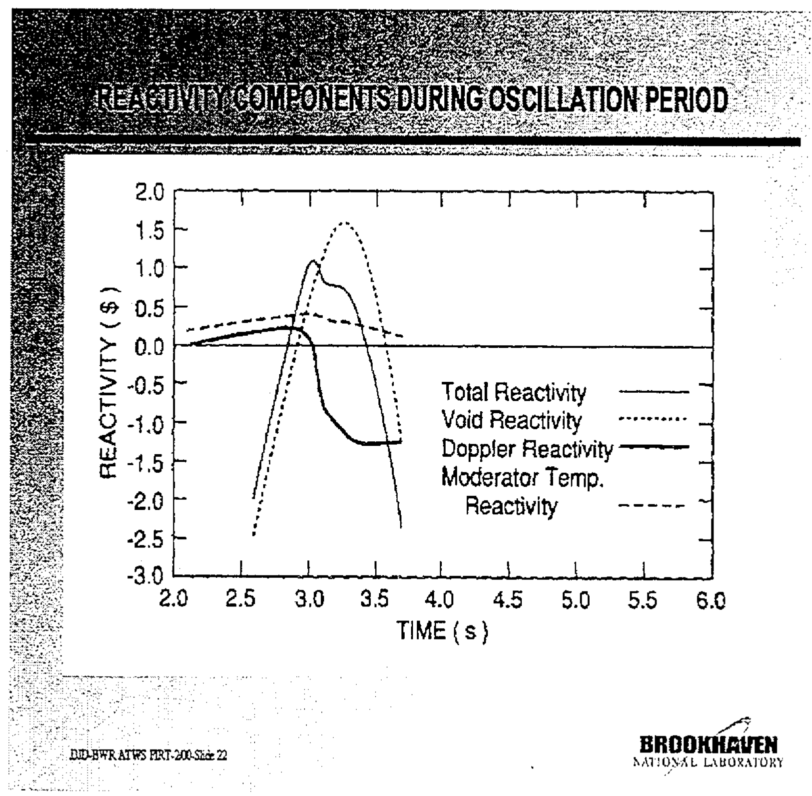
It gets above one dollar, albeit very briefly, and the average is less than zero, i.e., it's negative (see Slide 21).

Slide 21



It's also of interest to look at the details of reactivity within one cycle as increasing initially, reflecting a situation where voids are collapsing. With void reactivity increasing, total reactivity is essentially following the void reactivity. Total reactivity peaks just above one dollar, just above prompt critical, as a result of the Doppler feedback, because the Doppler feedback increases and provides a negative reactivity component. This causes the large peaks seen in the power traces. Total reactivity then continues downward as a result of the very strong void effect. During this portion of the cycle, the void fraction is increasing and, therefore, the reactivity is becoming more negative. The reactivity behavior is complex and both fuel temperature and void feedback have important effects.

Slide 22

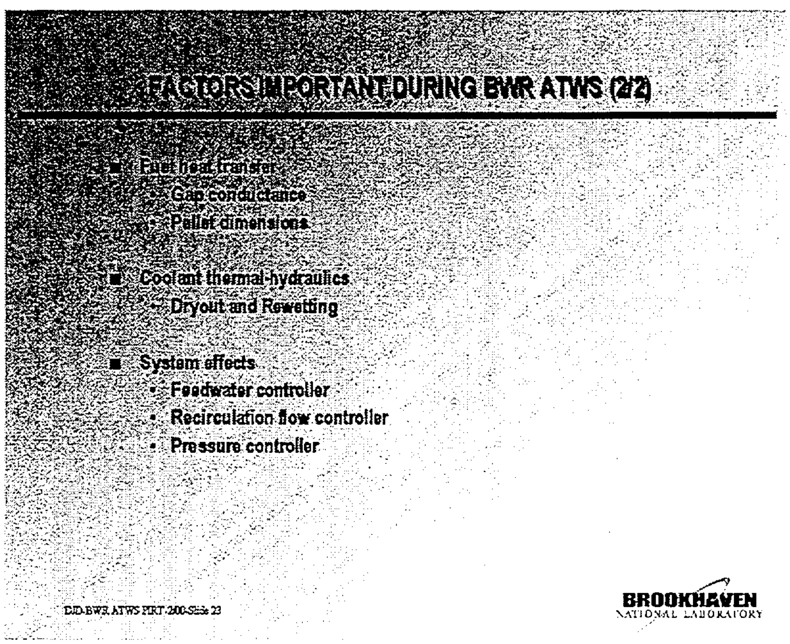


Slide 23 shows some other important factors. The fuel heat transfer is important and depends on gap conductance, pellet dimensions and other parameters. The gap conductance is a strong function of burnup. The pellet dimensions depend on whether we are talking about 8x8 fuel (as in LaSalle) or other configurations. Many different types of fuel bundles are used, and as the vendors go to smaller and smaller pins, the fuel pellet diameter changes and the heat transfer characteristics of the different fuel pins change.

Coolant thermal-hydraulics is obviously important, and dryout and wetting are highlighted because of their importance in determining fuel conditions.

I'd also like to mention system effects. There are various controllers in the system, and how they respond to a particular event is also important in determining the amplitude of the oscillations.

Slide 23



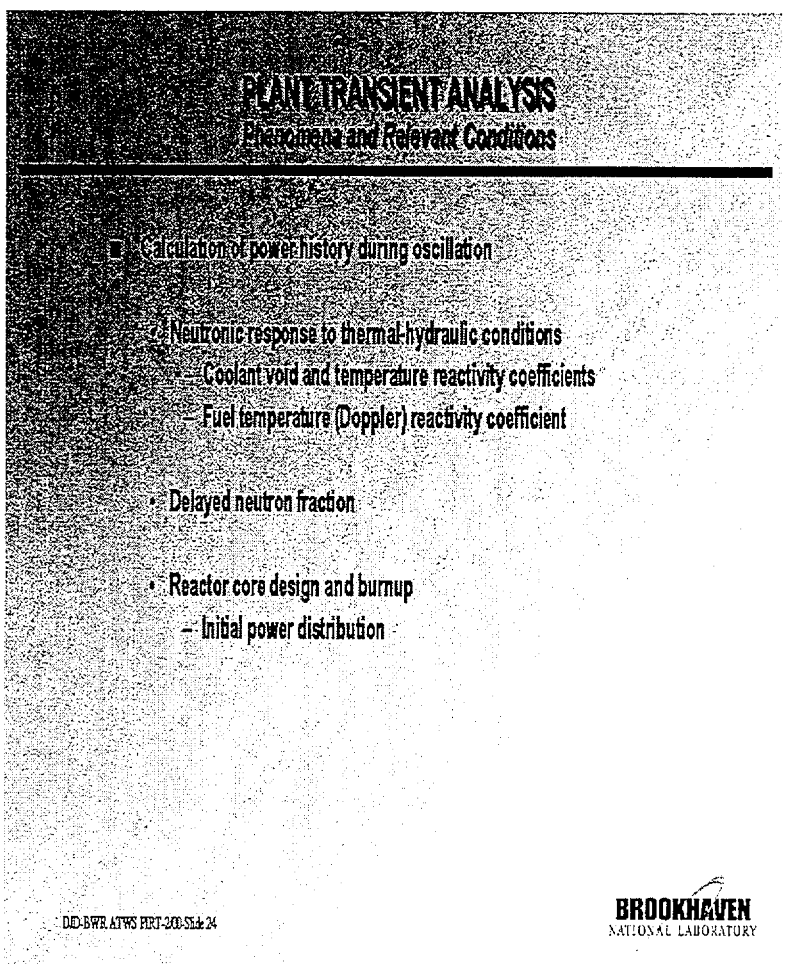
Phenomena and Relevant Conditions

Let me finish by presenting a table of phenomena and relevant conditions (Slides 24-26), which can perhaps be the starting point for the deliberations of the PIRT panel. I've broken it down into two elements, the first of which is the calculation of the power history during the oscillation.

On Slide 24 there are neutronic parameters. The neutronic response to the thermal-hydraulic conditions is important, and that includes the coolant void and temperature reactivity effects or reactivity coefficients. It includes the fuel temperature or Doppler reactivity effect or coefficient.

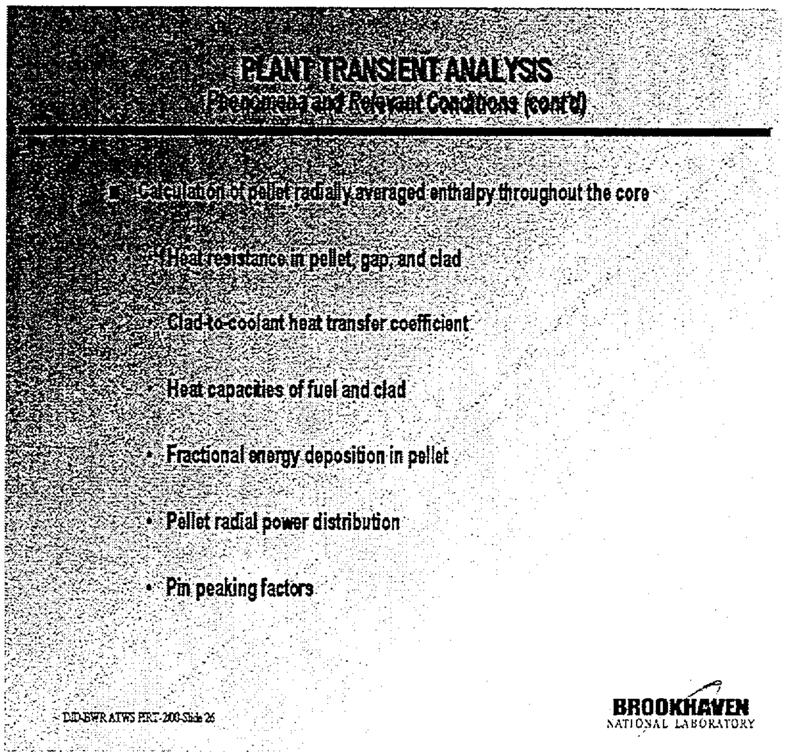
Delayed neutron fraction is another factor, which obviously impacts dynamic behavior and is affected by the burnup of the fuel. Reactor core design and burnup, as always, are important and imply, among other things, initial power distribution, which is important in determining the stability of the reactor and how the reactor will respond.

Slide 24



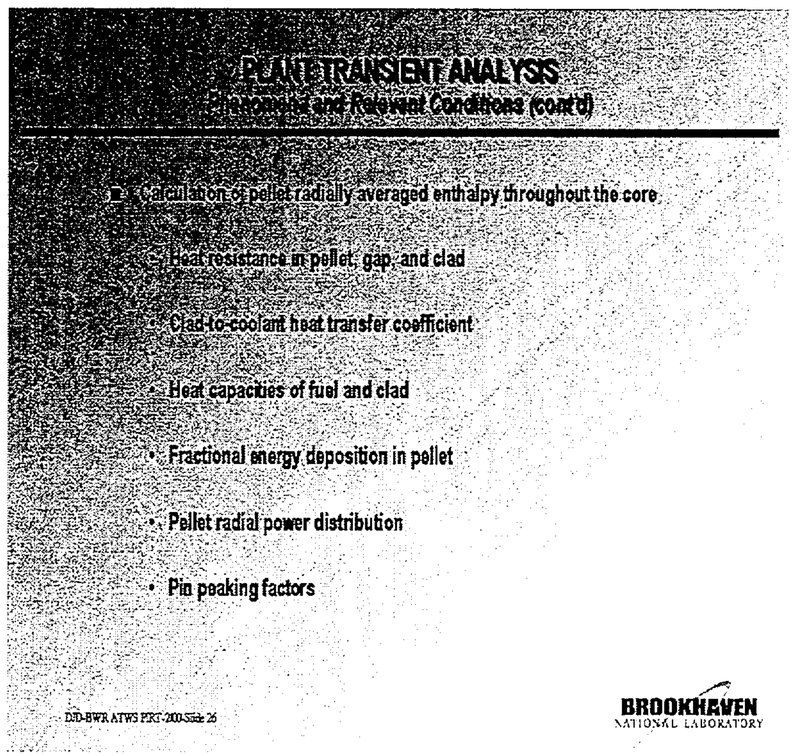
Thermal hydraulic behavior also determines the power history. The thermal hydraulic behavior determines the temperatures and the void fraction, which in turn determine power history. Slide 25 lists some of the items introduced earlier, which won't be discussed further.

Slide 25



The second element in the table of phenomena and relevant conditions is specifically the calculation of the radially averaged pellet enthalpy throughout the core by assuming we're still using that as the criterion for the bottom line which is the prevention of fuel dispersal and the maintaining of coolability. There are many things that go into the calculation of the enthalpy: heat resistance (obviously); the clad-to-coolant heat transfer coefficient, which raises the question of critical heat flux; heat capacities; fractional energy deposition in the pellet; and, the pellet radial power distribution. Pin peaking factors are important if you're not modeling every single pin explicitly, in which case pin peaking factors are used to get to local fuel enthalpy.

Slide 26

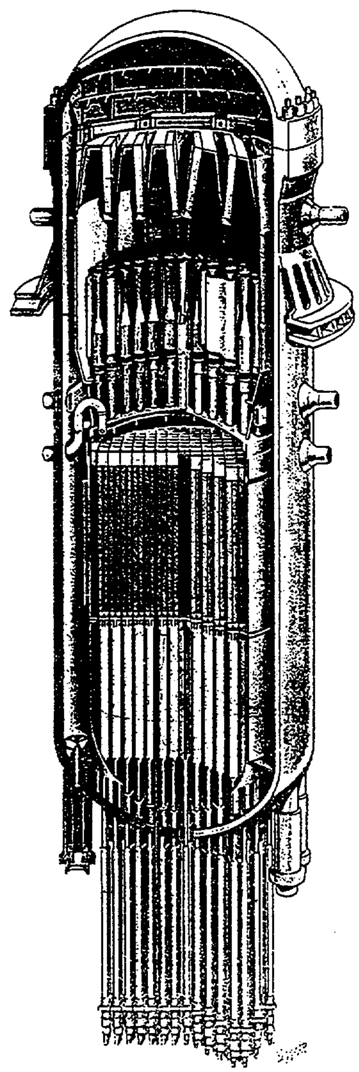


G-2. BWR ATWS CALCULATIONS

This review was prepared by Keijo Valtonen of the Radiation and Nuclear Safety Authority in Finland.

MR. VALTONEN: I have a lot of similar things in my presentation. The only difference is that I didn't use the reference plant in my calculations, because I didn't have any information from LaSalle. Instead I chose TVO-1 where some years ago occur an oscillation incident. I'd like to show you which kind of the boiler it is. It looks like that.

It is 20 years old, boiling water reactor, ABB design, and the difference between LaSalle and ABB reactor is that in ABB design, there are internal main recirculation pumps.



All the other components are similar to LaSalle. So I don't see any reason why these results are not also good for your purposes here, to make judgment of how serious this scenario can be.

First, I would like to show you what kind of ATWS scenarios we have chosen and analyzed during the last ten years.

BWR ATWS OVERVIEW
ATWS and oscillations

- MOST IMPORTANT SCENARIOS
  - MSIV CLOSURE
    - OVERPRESSURIZATION
    - FEEDWATER TEMPERATURE REDUCTION
    - WATER LEVEL REDUCTION
  - LOSS OF FEEDWATER
    - OVERPRESSURIZATION
    - WATER LEVEL REDUCTION
  - LOSS OF FEEDWATER PREHEATERS
    - FEEDWATER TEMPERATURE REDUCTION
    - POWER INCREASE

The main reason why we have done these ATWS analyses is that we have been interested in, not so much about oscillations itself but reactor over-pressurization and the mass inventory in the core and because the mass inventory --

MR. VALTONEN: Okay. And there are actually three different ATWS scenarios which we have chosen, MSIV closed, loss of feedwater, and loss of feedwater preheaters. The first ones are more important if we are analyzing reactor overpressurization, mass inventory and containment behavior.

But I think that the most significant case, if we are looking at oscillation scenarios, is loss of feedwater preheaters, because there is also a power increase due to reduction of core inlet subcooling at the same time when the plant is operating at low flow, high power condition. And that incident, which happened a few years ago in TVO plants, was exactly that kind of incident. So I will describe a little bit later that scenario.

Well, it's important first to know what kind of ATWS management strategies are used at TVO plants. The most important thing is try to reduce the reactor power. There are several different things which operator can do in order to reach that goal; either reduce the water level in the reactor or inject boron into the reactor, and also, in some cases, reduce the reactor pressure by automatic depressurization system.

BWR ATWS OVERVIEW

ATWS MANAGEMENT STRATEGY

- REDUCTION OF REACTOR POWER
  - REDUCTION OF WATER LEVEL
  - PCP TRIP
  - BORATION OF THE REACTOR (AUTOMATIC OR MANUAL)
  - SOME CASES ALSO AUTOMATIC DEPRESSURIZATION
- PCP TRIP DRIVES THE CORE TO LOW FLOW/HIGH POWER CONDITION - MORE UNSTABLE
- REDUCTION OF WATER LEVEL EFECTS ON NATURAL CIRCULATION FLOW AND MAKES REACTOR CORE MORE UNSTABLE

But these counter-measures effect rather slowly on the reactor power.
So these counter-measures are not enough.

And that's how operators are asking to do at the moment in ATWS procedures. What is important to understand also that each core loading has different stability characteristics and we have large uncertainty how fast core oscillations are diverging, because the stability characteristics are sensitive for many different kind of things, which are presented here.

BWR ATWS OVERVIEW

Reactor core stability

- EACH CORE LOADING HAS DIFFERENT STABILITY CARACTERISTIC
- SENSITIVE FOR
  - FUEL BUNDLE DESIGN
  - FORM LOSS COEFFICIENT IN THE FUELBUNDLE
  - TWO PHASE - ONE PHASE FLOW RATIO
  - POWER PEAKING FACTORS
  - CONTROL ROD DENSITY IN THE CORE
  - VOID REACTIVITY COEFFICIENT
  - CORE INLET ORIFICING
  - PRESSURE LOSS ACCROSS THE STEAM SEPARATORS AND UPPER TIE PLATE

It is sensitive for fuel bundle design and it is extremely sensitive for form loss coefficient and pressure loss distribution in the fuel bundle. Also oscillations are sensitive for pressure loss across the steam separators and upper plenum. If we reduce the pressure loss in these steam separators, the core is becoming unstable. Also, uncertainty in these form losses is about plus/minus 20%, even in some case plus/minus 50%.

The core loading is the one important because it determines the power peaking factors. It has presented in this slide. If we have a very high power peaking factor, the core is unstable, and if we have a bottom peak axial power distribution, it is also making the core less stable.

The basic problem with stability is in one phase and two phase pressure drop. The oscillations are starting from the bottom of the core. There are small perturbations, temperature perturbations, which are generating some kind of small oscillation in the boiling boundary, and then when it is propagating to the two-phase area, so the same situation is there, too, that the two phase flow are starting to oscillate.

MR. VALTONEN: If we decrease the one-phase pressure drop in the channels, we will get a less stable core. This is the basic phenomenon. So it is the so-called density wave oscillations are starting from the bottom of the core and going through the core by flow.

MR. HOCHREITER: The one thing -- maybe my analogy was backwards, but I thought that one of the things you considered in the boiling water reactor design was you made the inlet resistance into the fuel higher loss coefficient than you would make the exit resistance, and that was supposed to add, I thought, stability to that.

You're saying that if you decrease the exit loss coefficients, you go to a more unstable situation?

MR. VALTONEN: The inlet orificing is one of the most important things in all of this oscillations, so that if we have a strong orifice at the inlet of the core we have more stable thermal hydraulics flow channels.

Then fuel designs in different reactors are nowadays exactly the same. But there are differences in separator designs and also the upper plenum geometry is different. The pressure losses in that steam area is very important, as I mention already and is very important to model these components accurately and take uncertainties in the form loss coefficients into account.

MR. HOCHREITER: I'm not debating that. What I'm wondering is if the basic design is to have a higher inlet resistance for stability relative to the exit resistance, I thought you said that as you decrease the exit resistance, you go to a more unstable situation.

MR. VALTONEN: Yes.

MR. HOCHREITER: That seems counter.

MR. VALTONEN: I think that the best measure is --

MR. ANDERSEN: If I could just make a comment. I believe you made that comment about the steam separators and not about the top of the channel.

MR. VALTONEN: Yes.

MR. ANDERSEN: If you increase the loss at the top of the fuel bundle, you become less stable.

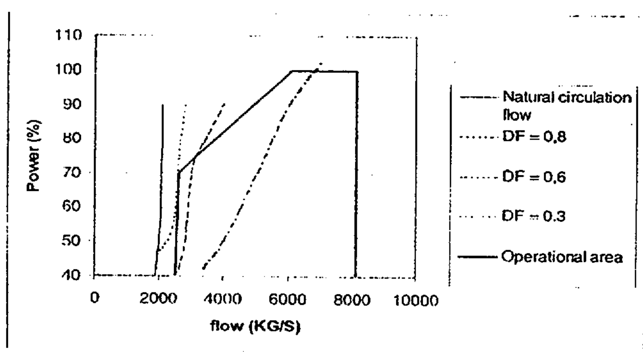
MR. VALTONEN: Yes. That is correct, yes.

MR. VALTONEN: And a very good measure to look at how stable the core is to use damping factors. Damping factors gives you information how fast the flow oscillations are diverging. If the damping factor is one, there is limited oscillations, and if it is above one, oscillations are diverging.

So I calculated damping factor for the reactor core just before that oscillation incident.

If you look at this figure, you can see that -- well here I have a natural circulation flow line and then I have the damping factoring along this line - here it is 0.8, in this it is 0.6, and here it is 0.3.

BWR ATWS OVERVIEW



The reactor is operating normally in this power flow map. So everything here is ok and damping factor is 0.3, it's very, very stable. But when main recirculation pumps trip and the core is operating, for instance, to this area, so these contour curve line are very close to each other. So oscillations are very sensitive for operating conditions.

And all the things are changing very rapidly in this picture. In this natural circulation flow line the damping factor in this case is a little bit over one, it's 1.02. But in TVO reactors, there is minimum speed for the pumps. So that minimum pump speed limits the pump core flow.

If you lose the power for the main recirculation pumps and the pumps are running down, then we have natural circulation flow conditions. But it is important to understand that each core loading, we have different stability characteristics. In some cases, the damping factor is almost 1.1. So the core in that case is really unstable.

If we inject cold water into the core, and water temperature is dropping down and this cold water is entering into the core, then the damping factor is becoming even higher.

MR. PEDDICORD: Keijo, as part of your comment, with a different core loading having a different set of situations, then is it correct this is also impacted by the extent of burnup in the fuel and it's changed if you have more highly burned fuel in the core?

MR. VALTONEN: If we increase the burnup in the core, so we are changing also the core loading strategy, because then we must put the high burnup, fuel bundles into the core center and there are also less fresh, more reactive fuel bundles in the core.

And in high burnup loading, these fresh fuel bundles are also more reactive, because otherwise it's impossible to run the whole cycle.

And the difference between high burnup fuel bundles and adjacent fresh fuel bundles is really big in terms of reactivity and power peaking factors.

MR. PEDDICORD: And so it makes the situation worse for this situation.

MR. VALTONEN: I this case, the maximum bundle average burnup is 40 MWd/kgU. So it is not very high. The maximum rod burnup is around 48MWd/kgU - 50 MWd/kgU.

MR. HOCHREITER: Do you generate one of these -- I think you said you generate one of these maps depending upon the fuel type.

MR. VALTONEN: Yes. In this case, we have nine-by-nine fuel, but we are doing these kind of calculations for each loading.

MR. HOCHREITER: For each loading pattern.

MR. VALTONEN: So that we can be sure that every loading pattern is stable or inside this box.

So this is some basic information from that case which I chosen for my calculations. Reference plant is Olkiluoto and the power is 2500 megawatts. There are internal Primary Coolant Pumps (PCPs).

BWR ATWS OVERVIEW

(5)

- REFERENCE PLANT OLKILUOTO 1
  - NOMINAL POWER 2500 MW
  - NOMINAL RECIRCULATION FLOW 7600 kg/s
  - INTERNAL PCP
- OSCILLATION INCIDENT AT 1987
  - INCIDENT OCCURED DURING STARTUP OF THE PLANT
  - BYPASS OF FEEDWATER PRE-HEATERS
  - DF 1.02 AND FREQUENCY 0.47 Hz
- TRAB 3D
  - 3D NEUTRON KINETICS
  - NONEQUILIBRIUM, NONHOMOGENEOUS THERMAL HYDRAULICS

Oscillation incident occurred almost ten years ago and the incident occurred during startup of the plant. Maintenance personnel did some tests for bypass valves of feedwater preheaters and they did some mistakes and then suddenly they bypass the feedwater preheaters permanently and cold water entered into the core.

In that case, the damping factor was 1.2, at the low flow, high power area, and the core power starts to oscillate and oscillation frequency was 0.47 hertz.

MR. HOCHREITER: Was the plant at full power or approaching full power? You say startup.

MR. VALTONEN: The plant was just starting up and operating point was in this power, low flow condition.

MR. HOCHREITER: Could you point that out on the previous slide, on the power map, whereabouts you were?

MR. VALTONEN: This one here?

MR. HOCHREITER: Yes.

MR. VALTONEN: It was here in this corner point.

MR. HOCHREITER: Okay. Right in the corner point.

MR. VALTONEN: And I used in this calculation our TRAB3D code, We have developed it last four years. First version was 1D, and now we have a 3D code. TRAB3D has 3D neutron kinetics and non-homogeneous thermal hydraulics. So it is a normal transient code.

MR. DUNN: What is the relationship between the spatial definition of the thermal hydraulics and spatial definition of the kinetics?

MR. VALTONEN: In our model we have 500 channels, thermal hydraulics and neutron channels.

So each individual channel has a description in that model. Each thermal hydraulics and neutronic channel has modeled, so it is exactly one-by-one. Then we have the controllers also in the model and all the main components are present in there.

MR. HOCHREITER: What is the slave channel on the figure?

MR. VALTONEN: It is actually the one channel which we use normally when we calculate critical heat flux type of calculations.

MR. HOCHREITER: So it will draw boundary conditions from whatever power assembly you want it to.

MR. VALTONEN: Yes.

MR. HOCHREITER: Okay.

MR. VALTONEN: Initial condition is presented here. So the power was 1300 megawatts, inlet flow is almost 3000 kg per second, and feedwater temperature is 163 degrees of C. And inlet sub-cooling was 21 degrees of C. This core bypass flow is important, it was 11 percent in that case. So the fuel which we used in these calculations was rather old-fashioned.

BWR ATWS OVERVIEW

- INITIAL CONDITIONS

• POWER	1300 MW
• INLET FLOW	2980 kg/s
• FEEDWATER TEMPERATURE	163 DEG C
• INLET SUBCOOLING	21 DEG C
• CORE BYPASS FLOW	11 %

MR. ANDERSEN: Let me just ask a question. These numbers here, those are typical of the low flow, high power conditions.

MR. VALTONEN: Yes.

MR. ANDERSEN: Eleven percent bypass fraction is typical of normal operation. Once you get into reactor instability, the bypass flow fraction will be substantially lower.

MR. VALTONEN: You are correct. It is a bypass flow during normal operation. And the model is taking this into account automatically. So that's why I didn't put the right number here, because it's already defined in the model.

One of the biggest problems in these oscillations is that we have very little experimental information available. So we don't have much possibilities to verify our codes. I can show you two different -- you can't see this one, it's not a very good slide.

This is a measurement from that incident. It's neutron power, averaged power range monitors

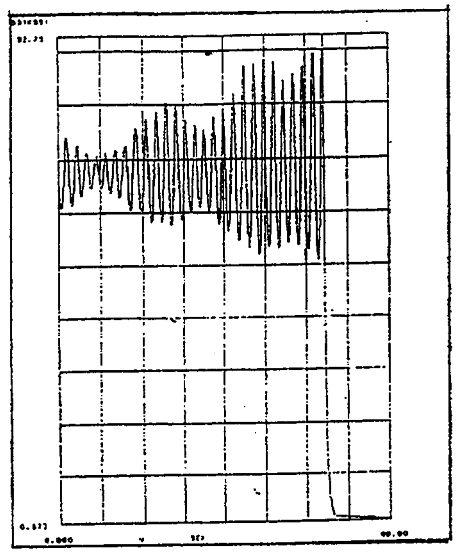


Figure 5. Fission power

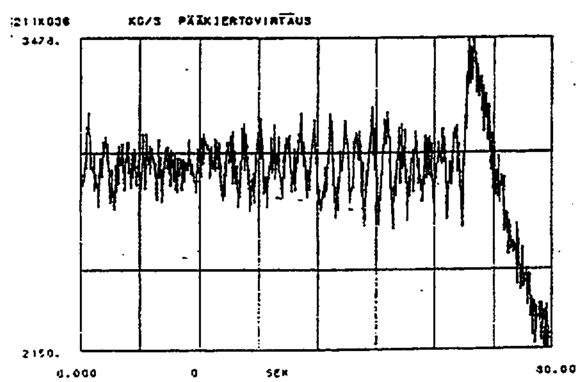


Figure 7. Main recirculation flow

So you can see here how the core power oscillates. You can also see that beat in these oscillations and this beat is coming from the fact that there are number of different channels, the oscillation phase of which is little bit different. So when we are talking about an in-phase oscillation, it is not exactly true that oscillation are at the same phase in each flow channel. The oscillations frequency depends on the channel power.

And what we can also measure, which is not actually very good one, is the main recirculation flow. There are in that reactor eight delta P measurements at the inlet of the core. So the measured flow an average over these eight measurements.

MR. MOTTA: Keijo?

MR. VALTONEN: But it is not a very accurate one. It's very difficult to use any credit.

MR. MOTTA: Could I ask you what the axes are on the two graphs? I guess it's time on the --

MR. VALTONEN: Starts at 2150, top is 3478. So the plotting routine is putting these numbers automatically.

MR. MOTTA: But could you tell us what you are plotting?

MR. VALTONEN: This is inlet flow into the core. So that measurement shows you that there really are some oscillations, but what is the amplitude in each channel is another story.

MR. MOTTA: And the abscissa is in time.

MR. VALTONEN: It's in seconds, it's from zero to 80 seconds.

MR. MOTTA: Is it the same on the upper graph?

MR. VALTONEN: Yes. It is from -- this is almost zero down here and this is 100 percent power. The scram will occur lower.

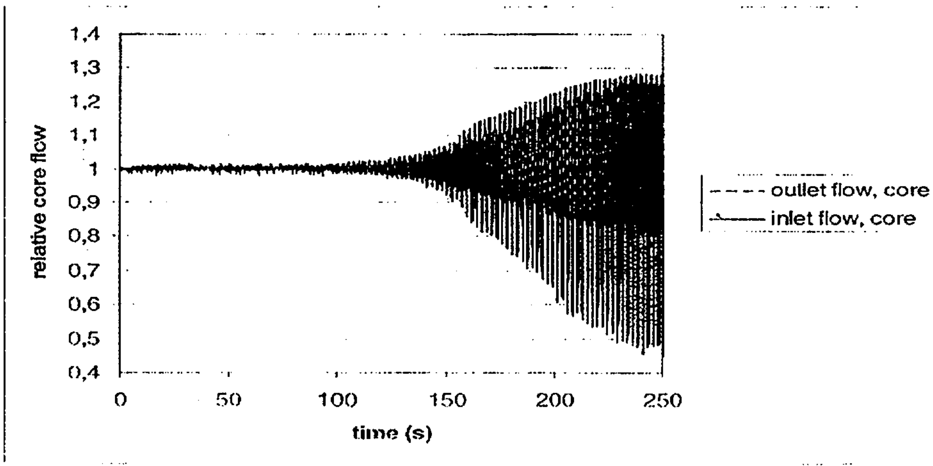
So we have a flow power scram line also. The scram is occurring in this point, which is 92 percent power. That scram line is just a little above this power flow map. The scram line is here.

So I used our code and tried to repeat these oscillations and now I didn't take into account the scram at all. So it looks very much the same. Here you can see the inlet temperature. So bypass at the preheaters increase the inlet sub-cooling and the power is starting to increase and after few second it starts to oscillate.

So I stop my calculation to the point when void fraction at the outlet was about one. So I thought that it is not very good idea to continue because some of the correlations are not valid in that area.

And then this core inlet and outlet flow oscillates. You cannot see very much in this figure, only that this inlet flow is going very fast to the zero. I did not continue the calculations further because outlet void fraction was one at the same time when flow reverse. But I think that in oscillation incidents the flow reversal at the inlet of the hot channel is an important phenomenon.

BWR ATWS OVERVIEW



MR. HOCHREITER: What does the pressure do, particularly if you have -- is there a set point that you control pressure to, so that if you would over-pressurize the system, a valve would open; if you'd under-pressurize the system, the valve would close?

MR. VALTONEN: You mean safety valves.

MR. HOCHREITER: Not so much safety valves. Either power operator relief valves or something like that. In other words –

MR. VALTONEN: In that case, the pressure controller is in the normal condition. It's keeping the pressure almost constant all the time.

There are, of course, some fluctuations because of these oscillations, but it is not very high.

MR. HOCHREITER: And how does it control pressure? By basically increasing or decreasing a resistance in a line?

MR. VALTONEN: Well, it is actually turbine control valves, which are keeping the pressure constant. So the controller is controlling turbine control valves all the time.

It is a quite accurate way to keep the power -- to keep the pressure constant, also, in this case. We measured also the pressure, but I didn't show you that pressure measurements because they are not very interesting.

MR. VALTONEN: But the MSIV closed case and all those over-pressurization case, there are power pressure oscillating between opening and closing pressure of the relief and the safety valves. So that it is like a pressure oscillation are different that in flow induced oscillations.

MR. TULENKO: Now, is this mass flow rate we're looking at?

MR. VALTONEN: Yes, it is the mass flow rate at the inlet and outlet. I will give you a little bit more detail later on that.

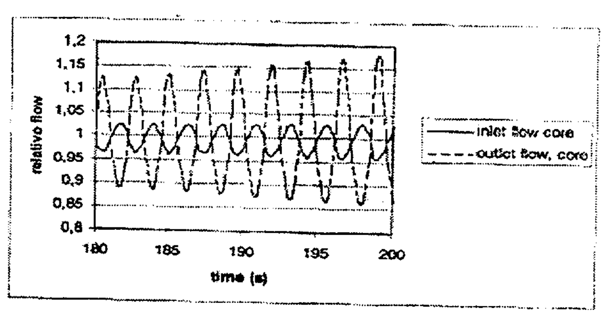
MR. DUNN: On that flow rate, that was the average channel?

MR. VALTONEN: Yes, that was an average channel.

MR. VALTONEN: Well, I show you also there are four average inlet and outlet flows and this is actually the inlet flow. There is an outlet flow oscillation. Here you can see that over all channels, these oscillation amplitude is not very high, because also those channels are included into this average flow, which are operating very low power level. Those are not oscillating at all or the amplitude is very low.

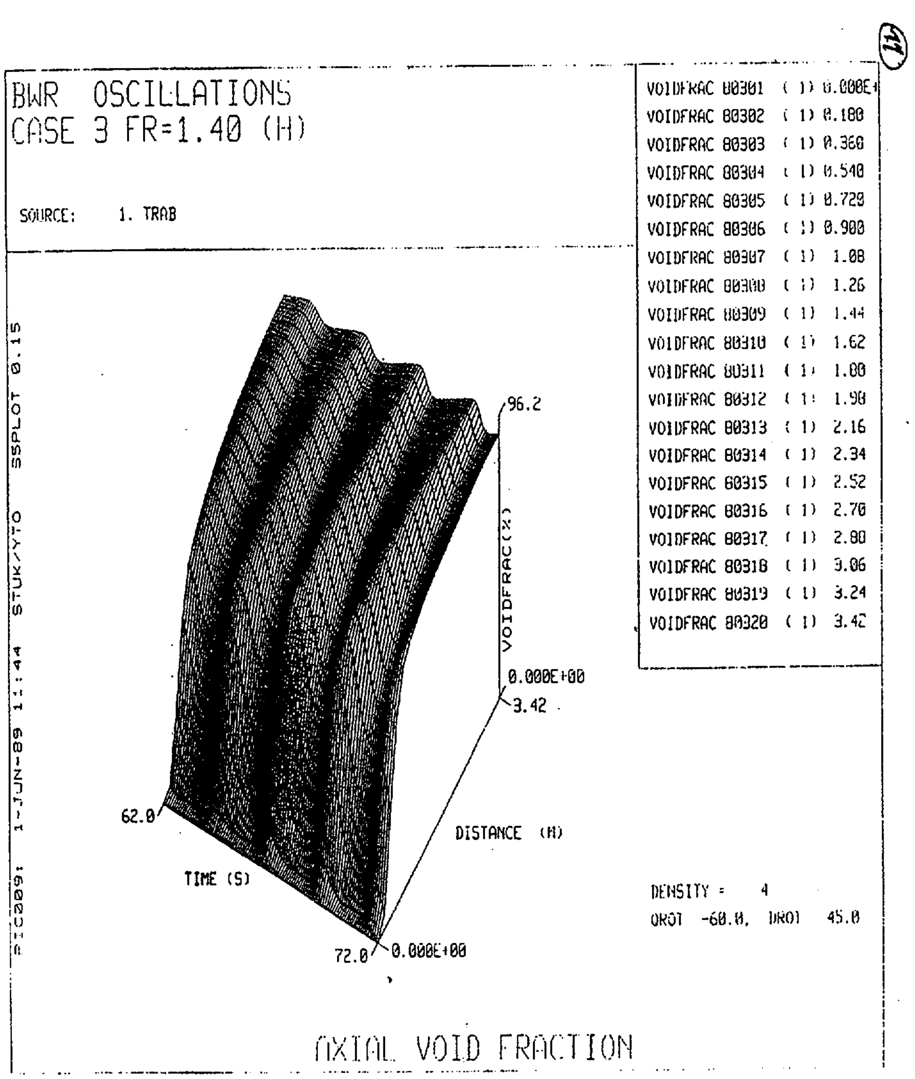
BWR ATWS OVERVIEW

core average inlet and outlet flow



So when we take a sum over all the oscillating channels, we reduce that amplitude very fast. But it is important to note in this case that there is 180 degrees phase difference between inlet flow and outlet flow oscillations, and that is important because when I have done these calculation, I have come to the conclusion that we cannot use point kinetics in these cases, because the thermal hydraulics is rather complex and it effects axial power distribution.

It is basically a question of density wave oscillations, and you can see how these density waves are propagating through the channel. If you look at 0.72 seconds here and there is a void fraction and this is from bottom to the top of the core, and the perturbation is starting from here. It is going through the core and coming here a few seconds -- one second later to the top.



So we have 180 degrees phase difference between inlet and outlet flow of the core. The flow is also oscillating 180 degrees phase difference and that is important also when we're looking at how the fuel is behaving. As I told, in these conditions normally, the axial power peaking factor is strongly bottom peak. In that case, the top is here and the bottom is here. The void fraction perturbation is starting from here and going through the core. You have a maximum power here at the bottom and minimum power here at the top.

MR. HOCHREITER: Can you again say where the top and the bottom are?

MR. VALTONEN: 3.42 meters is the bundle length.

MR. PEDDICORD: The bottom is the top.

MR. VALTONEN: Yes. Because it was really tricky to draw these 3d plots. If you turn this figure around, you cannot see anything. And this is the only way to give you some idea how these oscillations are evolving.

MR. HOCHREITER: Do you also see oscillations in the downcomer that are either in or out of phase?

MR. VALTONEN: I'm coming to that downcomer problem a little bit later, it is a problem. Especially in those cases, where flow is reversing.

MR. DUNN: You used the term flow reversal a number of times. What is the definition of relative flow?

MR. VALTONEN: It is zero in these plottings.

MR. DUNN: So relative flow is no flow.

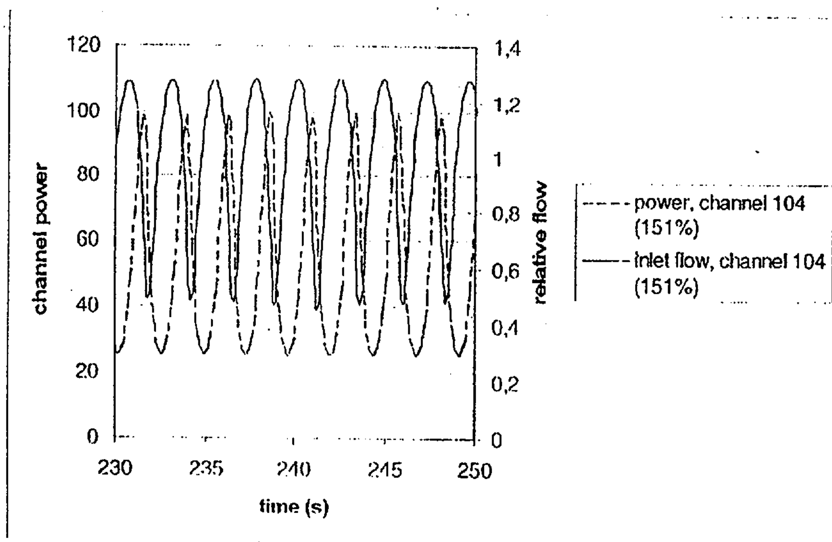
MR. VALTONEN: Yes.

MR. DUNN: So actually 0.95 means minus five percent of something.

MR. VALTONEN: Yes, something like that. And in this case, I have drawn two different sets. You can see how this power and inlet flow behave, this is an average flow over the channel 104. Radial power peaking factor is 1.51. Here you can see also how these flow oscillations are driving force in these oscillations.

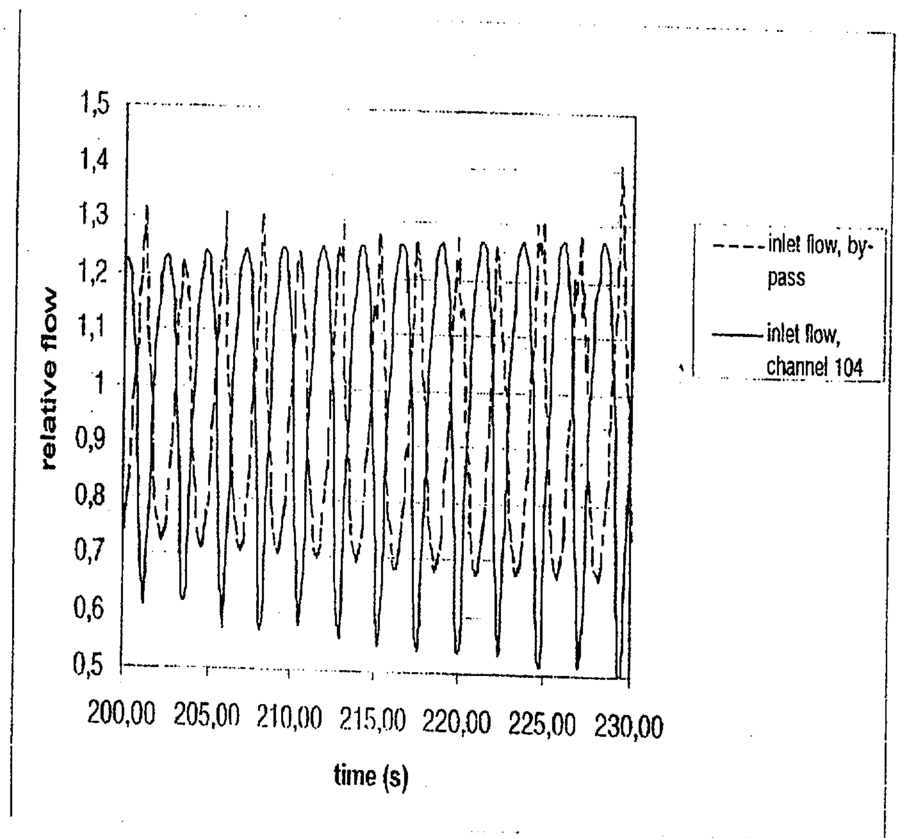
Flow oscillations are a driving force in these oscillations and power oscillations are coming just a little bit later than inlet flow oscillations into the picture.

BWR ATWS OVERVIEW



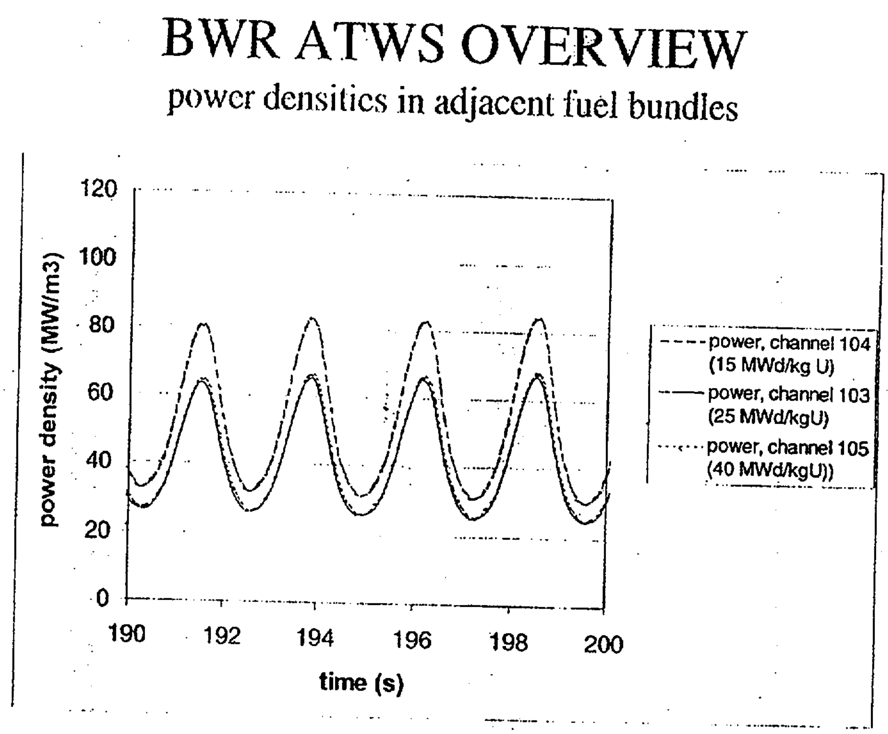
These only give you an idea how the different oscillations are evolving in the core. This gives you an idea how different channels are oscillating, depending on the power. I took into this plotting two different channels. The highest power channel and bypass channel. The bypass channel, we don't have any power, of course, in that case. I didn't put any heat into it.

BWR ATWS OVERVIEW



The oscillation phase difference between bypass and power channel is almost 180 degrees and the phase difference depends on channel power. High power channels are driving channels in oscillations at the beginning and then all the others are following them by small time delay. And when the oscillations are diverging, then also these little bit lower power channels are coming as driving channels. And it is important also to understand in that bypass is the very important channel, because it's like a counter-balance. So these oscillations must push this water column in the bypass channel upwards before the core starts to oscillate.

So in that case, I didn't put any energy into the bypass channel. That's why this is not a very conservative case, because in the real situation, we have some heat transfer into the bypass channel and there are also voiding in the bypass channel, which means that this counter-balance is lighter.



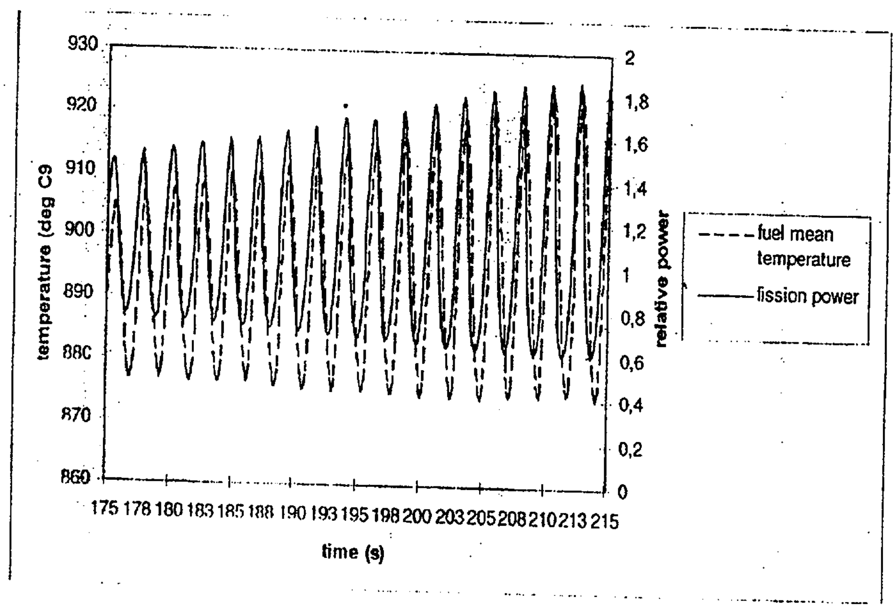
And because the bypass channel is becoming lighter, so the oscillations amplitude is diverging faster. And these are also important modeling aspects. We must have very detailed bypass channel model. And one of the big problems in the bypass modeling is that we have only one bypass channel in the whole core. Also, when we are using 3D codes. So it's a kind of one very huge bypass channel. And that is not correct, because bypass channel near by the high power are behaves differently than near the low power area.

Then I have some information how different channels are oscillating. The maximum power channel is the channel number 104 and that is 15 megawatt days per kilo uranium. And these two channels are adjacent to that high power channel.

And what I'd like to show you is no big difference between adjacent channels, Fresh fuel bundle and high burnup fuel bundle are behaving almost similar ways.

(51)

BWR ATWS OVERVIEW



And then I have a fuel mean temperature and fission power in this figure. This gives you an idea how fuel mean temperature is following the fission power oscillations. This is the mean fuel temperature in the channel.

The maximum temperature difference, peak to peak, in that point is 50 degrees of C. But if we look at some higher power channels, where the damping factors are much higher, then we have a much higher fission power spike and also the amplitude of temperature oscillations in the fuel is much higher.

And then I calculated also the fuel cladding temperature. It takes only seven seconds after reactor scram when fuel is experience boiling crisis. So the fuel cladding temperature is increasing very fast. The fuel cladding temperature reaches 800 degrees of C, before the termination of calculations.

The temperature anyway is increasing to 800 degrees of C. But there are no rewetting.

MR. HOCHREITER: Your rewet is locked out of the calculation.

MR. VALTONEN: Well, I look at those rewetting criteria and condition did not exceed limit where rewetting is possible.

MR. BOYACK: Your code would have permitted it, if conditions had --

MR. VALTONEN: Yes. If condition is correct, there is rewetting.

MR. HOCHREITER: What rewet temperature would you typically be calculating? What I'm worried about is that you may calculate that and you may calculate it because the correlation you have gives you a critical heat flux of zero when the flow goes to zero.

MR. VALTONEN: Yes, that's right.

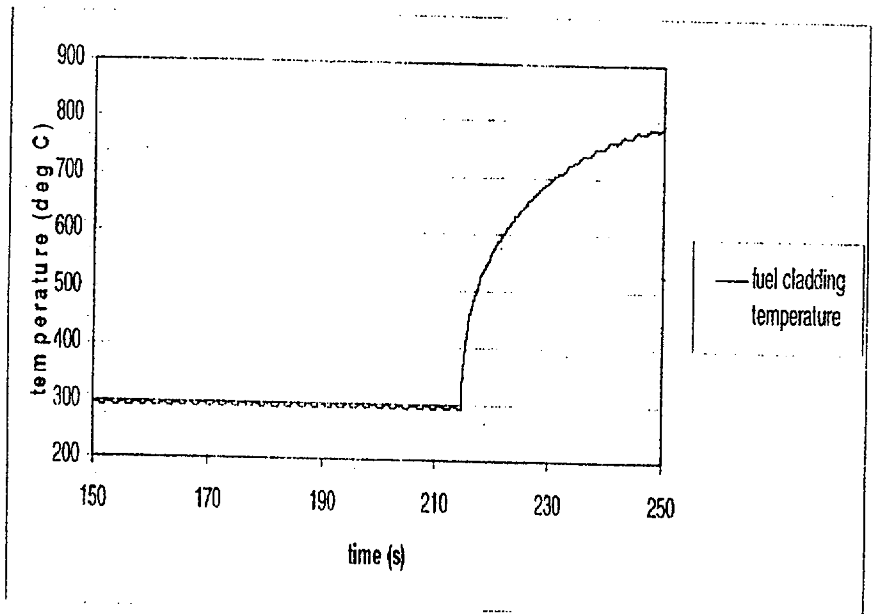
Mr. HOCHREITER: All right, So this may be an artifact of the calculations.

MR. VALTONEN: Yes, it is, It is. We actually cannot do that calculation using present DNB and CPR correlations, I am quite sure about that.

MR. DUNN: Larry, don't you think that that is matter for kind of the committee to decide here, as to how sensitive the results are to that, I'd like make one comments. This almost makes me wish we were having this presentation and this discussion after out next PIRT, because if I believe that perhaps the LOCA type general slow heat-up of the cladding changes the cladding characteristics a little bit, then we're into -- if we get the dryout -- a phase in which we're going to go into the phase change for the cladding and we're going to change the cladding characteristics and its actions with the fuel pellet, whereas before that, we've got the REA type of example.

I suppose it's a little too far to say we don't have to concern ourselves after we start that dryout heat-up. Maybe we ought to -- people ought to think about that a little bit and see whether we could throw that portion of the transient away. I know, Larry, you're not going to like that.

BWR ATWS OVERVIEW



MR. HOCHREITER: Well, no. I think you at least want to discuss it, so you can -- I would think you'd want to have some kind of a way of ending this thing, all right, and recovering the core.

MR. DUNN: True, but to say in line with Ralph, I think we're primarily hearing what are the possibilities for pellet-clad interactions or clad interactions that are bad with exposure and stuff like that, as opposed to systems.

MR. HOCHREITER: Yes, but it's all predicated on getting CHF.

MR. DUNN: The system is important because it sets up the possibilities that the cladding may have to go through.

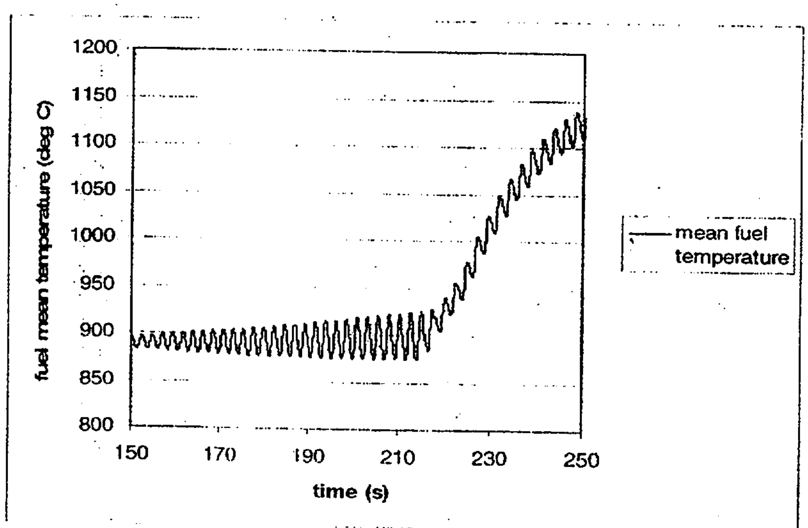
MR. HOCHREITER: Yes, And I'm afraid that this is an artifact of this calculation.

MR. DUNN: Could be.

MR. BOYACK: Why don't we go ahead and return to that after we have more of the presentation?

MR. VALTONEN: So this is fuel average temperature in the hottest channel. This gives you an idea what is happening there. So if you don't take into account this dryout here, so the oscillation amplitude is only increasing as it happened in the LaSalle case. But we have a dryout, so the temperature is jumping up very fast.

BWR ATWS OVERVIEW



Then we have -- we still have oscillations. So we have a high temperature and oscillations in the temperature. So this gives an idea of how the pellet is heating into the cladding all the time.

MR. HOCHREITER: In your calculation, when you have a flow stagnation, do you automatically go through critical heat flux?

MR. VALTONEN: It is very tricky condition, because all the correlations which I had used, I have used several different correlations in these calculations, CPR and DNB correlations, because actually when we look at these axial power distributions --

MR. HOCHREITER: Most of these correlations don't go to zero flow.

MR. VALTONEN: If you look at these oscillation conditions, alpha of the core is here and power density -- and we have a high void fraction here. So we have CPR conditions in that area. At the same time, we have a very high power densities in a sub-cooled area. So it is more or less like a DNB type of problem than CPR problem.

But if we have a reverse flow here, so all the correlation is -- we cannot trust anymore them. And the flow from this is also -- also the problem. All correlations, which we have, have been done for positive flow conditions.

MR. HOCHREITER: Right.

MR. VALTONEN: And that's why I don't believe those correlations at all. We must have some other type of correlations which we like to predict the incidents correctly.

MR. HOCHREITER: See, this situation, to me, is very much like gravity re-flood situations.

MR. VALTONEN: Yes.

MR. HOCHREITER: And if you run -- we would, like in the FLECHT experiments, we'd run them past quench, so you remove all the stored energy, and then you would get these density wave oscillations and the flow would vary, the void fraction would vary with time, and, in a PWR case, the thing that was driving more the oscillations than anything else was the heat release in the steam generators, because as you would push the two-phase mixture into the generators, you generate vapor and then you have a high resistance downstream, which is the pump.

So this would push back on the mixture level in the core and what you would see is the core level drop out, the downcomer level would come up, and then you'd burn out this steam. But we would never go through critical heat flux. And then the downcomer level drives the flow back onto the core and you start the oscillation all over again.

And I think you've got downcomers in your bypass regions, plus your own downcomer.

MR. VALTONEN: Yes. But if you look at these oscillations, it puts more and more power into --

MR. HOCHREITER: well, that's one thing that was not simulated in those tests.

MR. VALTONEN: And if you look at this case, it's interesting that the maximum void fraction at the outlet, the maximum power in the outlet is almost 180 degrees phase difference.

There is also CPR in the top and then after one second, DNB at the bottom. I have seen also these type of calculations --

MR. DUNN: I know, I'm just asking him what is accomplished.

MR. VALTONEN: Reactor scram and main recirculation pump run-down or trip.

MR. DUNN: So what we're seeing at 215 or 212 is the loss of flow to the core. That's why we're getting the departure from boiling and heat-up. Okay. Thank you.

MR. HOCHREITER: Yes, Now, you still have oscillations occurring in the core and you still have oscillations occurring in the hot assembly, don't you? So that you're getting water that's carried all the way up to the top of the fuel assembly.

MR. VALTONEN: Yes, We can see all these phenomena are continuing, all those phenomena are going on, although we have a dryout in one channel, because first dryout is in the highest power channel. Then it is escalating little by little to some other channels, when the amplitude of the power is increasing.

So these oscillations are diverging all the time. So suddenly we have a huge amount of bundles in the boiling crisis and if we don't have a rewetting, we know what happens then.

MR. HOCHREITER: What I was driving at is that you're -- are you still -- when the code is calculating that you dry out, are you changing somehow the fluid conditions in that channel?

MR. VALTONEN: Yes.

MR. HOCHREITER: To make it different than the adjacent channel.

MR. VALTONEN: It is changing automatically, of course, because the models are there, but are those models correct in that case, that is the other question.

MR. HOCHREITER: Well, the question I have is you still have a boundary condition that is determined by the system.

MR. VALTONEN: Yes.

MR. HOCHREITER: The inlet boundary condition is more of a systems response and you should be throwing water up and down these channels, even if you calculate dryout. So in a situation like this, I would think if your rewet point is 650 degrees C, you would have had water in contact with that rod before you reached T-min. So you should have rewet. That's what I don't understand, unless, for some reason, in this channel, there is simply not water.

MR. VALTONEN: Well, maybe that is the situation. I don't know, because it's out of the -our information at the moment. We don't know exactly what is going to happen after these kind of situations. There might be some damping.

MR. DUNN: Don't we have a core outlet void fraction up in the 90 percent range after 210 seconds or something like that Isn't it really relatively high? The upper plenum.

MR. HOCHREITER: Not core average, I don't think. He indicated that the hot assembly had a high void fraction.

MR. DUNN: I have got 96 going up to something like that, Larry, for the outlet. If there's not much -- that's at 72 seconds. There's not much up there to fall back into each one of these individual assemblies.

MR. HOCHREITER: It's being driven from below. The flow end of the assembly should be driven from below, unless you've done something to change the pressure drop characteristics of that assembly.

MR. DUNN: I don't see why you couldn't have avoiding in the middle of the assembly of 100 percent pretty easy, I guess.

MR. HOCHREITER: I'm not saying you can't, but I'm saying every two seconds, you're throwing water up there.

MR. DUNN: How far it is going?

MR. HOCHREITER: I think it's probably going out of the bundle, but that's what we have to find out.

MR. GOODWIN: Ed Goodwin, NRC.

MR. BOYACK: Come up to a microphone.

MR. GOODWIN: When I look at the relative flows and they're almost totally out of phase, the inference there is that total core flow remains more or less constant. And if that's true and it also appears that void fraction is remaining relatively constant, integrated over the entire outlet, then it sounds like the same amount of power is coming out time-averaged, and this doesn't make sense. Yet, we're predicting the condition is getting worse in the hot channel.

It seems to me that what's happening is the hot channel is moving around.

MR. VALTONEN: But more important is how the flow is oscillating at the inlet and at the outlet. So it's quite sure that although we have a constant power, we have a dryout, also. And if we have a reverse flow condition in some part of the channel, that is also the case.

So it is more dependent on flow conditions than power conditions in the channel, because the power -- well, it's quite clear that power is constant, except that it is increasing a little bit in that case, because the feedwater temperature, inlet sub-cooling is increasing all the time.

So that's why the average power is also increasing. But the oscillations, if we don't have that inlet flow, the inlet sub-cooling changes. We have a constant power if we integrate it over these oscillations

MR. DUNN: I know, I'm just asking him what is accomplished.

MR. VALTONEN: Reactor scram and main recirculation pump run-down or trip.

MR. DUNN: So what we're seeing at 215 or 212 is the loss of flow to the core. That's why we're getting the departure from boiling and heat-up. Okay, Thank you.

MR. HOCHREITER: On the next shot that you had, the next slide.

MR. VALTONEN: This one.

MR. HOCHREITER: Yes. Now, you still have oscillations occurring in the core and you still have oscillations occurring in the hot assembly, don't you? So that you're getting water that's carried all the way up to the top of the fuel assembly.

MR. VALTONEN: Yes. We can see all these phenomena are continuing, all those phenomena are going on, although we have a dryout in one channel, because first dryout is in the highest power channel. Then it is escalating little by little o some other channels, when the amplitude of the power is increasing.

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MR. DUNN: Don't we have a core outlet void fraction up in the 90 percent range after 210 seconds or something like that Isn't it really relatively high? The upper plenum.

MR. VALTONEN: It's something like -- well, if I remember correctly, about 600, I'm not quite sure about that number, because it's inside the code and I have not looked at it. I don't remember the number.

MR. MOTTA: Keijo, a question. The calculation, I think you said you start at 250 seconds because that's where the void fraction went to above one or something like that.

MR. VALTONEN: Yes, that's right. Instabilities are increasing very fast, because these correlations are also -- are not valid anymore in that area. So we lose the validity of some correlations.

MR. MOTTA: Do you anticipate that there would be anything that would get around this temperature or would the fuel eventually melt or the cladding melt?

MR. VALTONEN: I'm not quite sure about that, because we don't have any test material from this kind of situation. So that it is quite unclear in this case. We don't know.

MR. MOTTA: Because at 250 seconds, likely the operators would have intervened by this point, correct? No?

MR. VALTONEN: No way, I don't believe, because it's totally impossible.

MR. MOTTA: How long would it take for them to --

MR. VALTONEN: At least two minutes. I believe at least ten minutes before anything happens. We have tested that type of case.

MR. DUNN: Is that because the operators don't have any signals that tell them something is wrong?

MR. VALTONEN: Well, they can quite easily see that there are oscillations. So they can push a button. They can try to start up the main recirculation pumps. They can trip the feedwater pumps or son on. But these are the only things which they can do.

But the main problem is that operators must do their decision based on some information which they have and whey they have a transient, all the alarms are plinking all over control room, and they must first look at what are the most important ones.

MR. DUNN: I got confused a little bit. You said scram occurred at what time?

MR. VALTONEN: I'll show you this. The power is --

MR. DUNN: There is no --

MR. VALTONEN: Here at this point.

MR. DUNN: About 200 -- about 190 seconds.

MR. VALTONEN: 180 seconds, yes. It's a question of 20-30 seconds.

MR. DUNN: And what did that scram accomplish?

MR. BOYACK: That is ATWS.

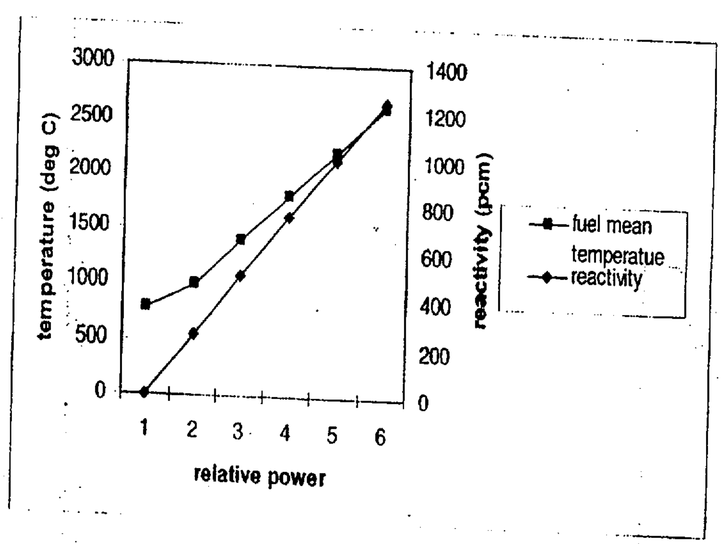
MR. HOCHREITER: Could you define for me mean fuel temperature? Is that core average, radial average across a given segment, at the hot spot radial average?

MR. VALTONEN: In our 3D code, we use a bundle as a one, it's an average bundle. If you like to get a maximum fuel temperature in that case, power peaking factor is something like 1.15.

MR. HOCHREITER: So this is at the highest -- at a high temperature elevation for the hot bundle.

MR. VALTONEN: Yes, And then I give you the timing of what happened in the different fuel. Notice the dryout and fuel temperature starts to increase after a while and also the center line temperature starts to increase here. This gives you an idea what is the timing of different kind of things. The fuel time constant.

BWR ATWS OVERVIEW



This is relative power, the maximum power during the oscillations. This is the temperature, fuel temperature, and this is the reactivity. I have taken this number from several different type of analyses and I think that it gives an idea of what is the melting point. So you must have six times original power, 600 percent power before you reach that condition. And the prompt criticality point is here. It's 500 PCM or something like that.

BWR ATWS OVERVIEW
UNCERTAINTY

- DNB AND CPR CORRELATIONS
  - LOW FLOW, STAGNANT AND REVERSE FLOW CONDITIONS
- REWETTING AFTER BOILING CRISIS
- CORE BYPASS CHANNEL BEHAVIOUR
- REVERSE FLOW
  - FORM LOSS COEFFICIENTS
- MODELLING OF LOWER PLENUM
- NOT ENOUGH EMPIRICAL INFORMATION FOR VALIDATION OF THE CODES

So I'm coming to these uncertainties in these calculations. So we have these DNB and CPR correlation problems, of course. We don't have correlations which cover these conditions. We have low flow, stagnant and reverse flow conditions. These are the biggest problems, I think, in this whole ATWS oscillation scenario, how to take into account these kind of things.

And rewetting is after boiling crisis, if there are boiling crisis. How fast does rewetting really happen, because we have 60 percent power in the core all the time. So this really high heat flux is coming out of the fuel if there are boiling crises.

So I don't know under what conditions, if it is exceeding that boiling crisis point, it cannot rewet anymore. So it's high heat flux in the core.

But we need some -- so I don't know what we need actually. We need new correlations, new type of models and so on.

MR. DUNN: One more question for you. In your rod model, did you have any cladding geometry change aspects model or did the --

MR. VALTONEN: No. No, it is not the fuel behavior model at all. It is the keeping the geometry constant all the time. It is quite normal 3D kinetics thermal hydraulics code.

MR. DUNN: Thank you.

MR. VALTONEN: This core bypass channel behavior is an important factor, also, because in the model fuel designs, we have a very huge bypass channel. First of all, we have a normal bypass and then there are also these internal water channels. In that low flow, high power area, during normal operational condition, there are already steam in these bypass channels.

What happens is we push more and more power into these channels, so it is important to note, because this bypass channel is getting lighter all the time if it is steaming, and especially the bypass channel is very sensitive for flow reversal.

The bypass channel flow is reversing first and then other channels. When the flow is reversing in the bypass channel, then it is also starting to boil. It's getting lighter and lighter. And because the bypass channel is counter-balance for the oscillations. So power oscillations are reversing much faster in that case.

MR. HOCHREITER: Now, the downcomer also oscillates. Does that oscillate?

MR. VALTONEN: I'm coming to that.

MR. HOCHREITER: Okay. I can't wait.

MR. VALTONEN: Yes. This figure shows what king of geometry we have used in our model. One problem in modeling is that from coefficients are measured in the normal flow direction.

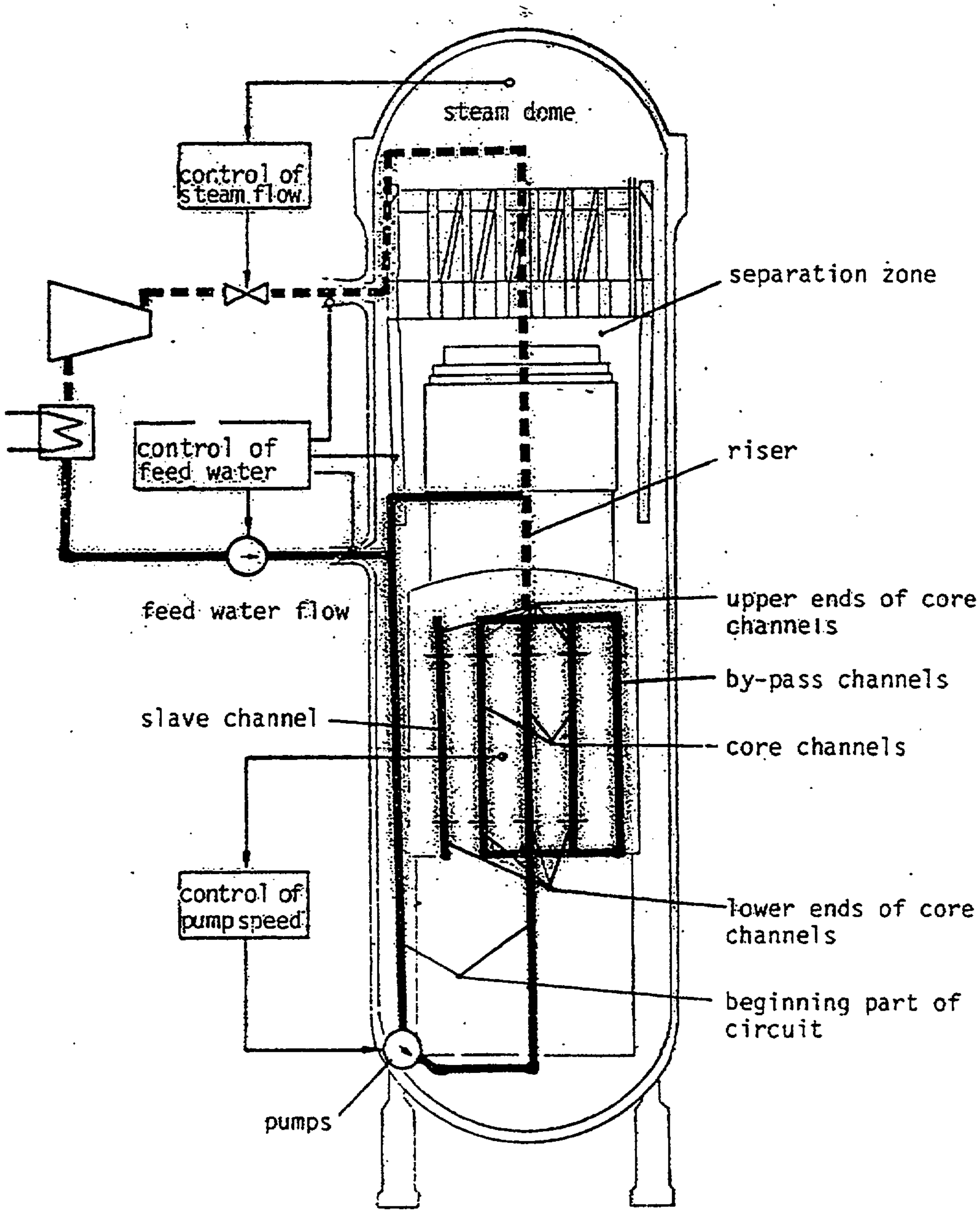


Fig. 1 Geometry of the system

So now in the modern fuel bundles, we have all kinds of fancy things, in the spacer, for instance, some kind of swiveling types of spacers, which throw the water through the cladding. And these spacers are very small form loss coefficient in normal flow conditions, but I believe that in the reverse flow conditions, those loss form coefficients are totally different area, much higher. This is the big problem, how to model these type of things into the code.

Well and now I am coming into the modeling of lower plenum, I'll show you how it is done in our model here. We had a common point here at the bottom for all the adjacent channels, 500 channels. So if we have a flow oscillation here, I'm quite sure that we need a better description of lower plenum, because there are some kind of the oscillations between adjacent channel through the lower plenum and bypass channel. This is also the case in all other codes, too.

So that is important to understand that there must be some kind of a 3D -- at least 3D model.

MR. HOCHREITER: Right now, your 500 channels goes to one?

MR. VALTONEN: Yes. It is one-dimensional loop.

MR. HOCHREITER: Okay.

MR. VALTONEN: And then it is only going into the one-dimensional -- 500 one-dimensional thermal hydraulic channel. And what I have done here in this, I tried to look at how important these lower plenum is and I did some calculation using the 3D thermal hydraulic codes, and I believe that it is very important to have that type of description of the lower plenum, so that we model it, because there are some kind of damping.

MR. HOCHREITER: Yes. The other thing I was getting at is if you have -- I mean, does the downcomer oscillate in phase or out of phase with the bypass? Which?

MR. VALTONEN: Excuse me. I didn't --

MR. HOCHREITER: Does the downcomer oscillate in phase or out of phase with the bypass?

MR. VALTONEN: It's out of phase.

MR. HOCHREITER: It's out of phase.

MR. VALTONEN: If I remember correctly. You mean bypass the normal flow channel.

MR. HOCHREITER: No. I'm saying the downcomer.

MR. VALTONEN: The downcomer. I don't know exactly how it behaves.

MR. DUNN: Why wouldn't you expect it to be like the bypass?

MR. HOCHREITER: I would.

MR. DUNN: Yes, okay.

MR. HOCHREITER: I wanted to get that confirmed.

MR. ANDERSEN: The bypass flow doesn't oscillate an awful lot. So the downcomer is more to the core flow than to the bypass flow.

MR. HOCHREITER: So it's really the downcomer that's the forcing function for the boundary condition at the bottom of the core, not the bypass. All right. That's Where I was at.

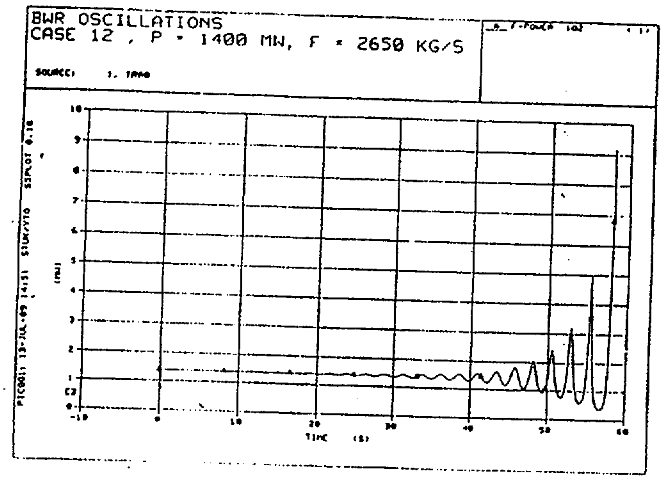
MR. VALTONEN: And my conclusion from these things are we don't have very -- we don't have enough test material so that we can know how exactly this fuel is behaving after the scram.

I'd like to show you one thing which is here also in my slides. This no bypass means that I have a voiding in the bypass. The bypass is in the model, but it is voiding. And I'd like to show you the model, because it gives you an idea how important this bypass channel is in these conditions.

You can see how fast this power is diverging in that condition, because we don't have a very strong counter-balance anymore. There are some water in the bypass channel, but there are also void in it in that case. That would be also in the megawatts here that we can see it.

BWR ATWS OVERVIEW

NO BYPASS



Okay. That was all I --

MR. HOCHREITER: Is there a way that you can do a mass balance and you know what the flow coming into the lower plenum is and you know what the flow coming into the fuel assemblies are and you know what the flow leaving the bypass region should be. Is the bypass a significant contribution to the total flow into the core?

MR. VALTONEN: No, not in that low flow area, it is not.

MR. BOYACK: Let's take a ten-minute break.

MR. DUNN: Can I just ask, first? You've indicated that you thought that the three-dimensional or at least two-dimensional modeling, in your case, was necessary because you showed us a bypass flow that was out of phase with the hotter channel flows, and then you also said that the low power assemblies were more in phase with the bypass flow than they were with the -- do you have a power density map per bundle or some idea of what that would look like, so we could judge how much of the power is going -- is in high powered assemblies and how much is in low powered assemblies?

I can think of a variety of ways to construct that and I don't even know whether it's uniform across BWRs or not.

MR. HOCHREITER: Or just what the relative power is between the outside and center.

MR. DUNN: I don't think -- I don't know -- that's not true for PWRs. Is that going to be true for a BWR? Are they more consistent relative to outside to inside?

MR. VALTONEN: I have some plots from that, too. The power distribution initial condition.

MR. DUNN: One thing you could say is it's just all over the map. If I answered that question on a PWR, I'd say forget it. It depends on what condition your vessel is in.

MR. VALTONEN: I think that there are not big difference between these two type of reactors, the power distribution looks exactly the same.

MR. DUNN: I'd like to see the percent of the core that's at various power levels.

MR. VALTONEN: Okay. I can do that.

MR. DUNN: Or some idea of that.

MR. VALTONEN: I have those plots somewhere.

MR. BOYACK: Okay. Ten minutes and then come back.

G-3. BWR ATWS Tutorial

This review was prepared by Jens G. M. Andersen of Global Nuclear Fuel (formerly General Electric).

BWR ATWS Overview
ATWS-Stability

J. G. M. Andersen

February 2000

The following presentation is an overview of some of the BWR ATWS issues. In particular, what is called ATWS stability will be discussed. A number of points in my presentation have already been covered and I'm going to cover those relatively briefly and a couple of other areas will be discussed in detail.

The presentation is covering the ATWS stability events and the licensing requirements (See slide 2). It will discuss the ATWS emergency procedures and what the phenomena and actions are that controls and terminates the event. There will be couple of examples on ATWS scenarios. Many of these examples originate from work done in the early nineties, after the LaSalle instability event. At that time there was a large effort involving the BWR Owners Group (BWROG) to evaluate the ATWS stability issue and come up with counter-measures to the ATWS instability event. Finally, the presentation covers some of the thermal hydraulic conditions that exist in the core during this ATWS instability event, because those are the boundary conditions that control the fuel behavior.

BWR ATWS Stability Overview

GNF

- Stability and ATWS Licensing Requirements
- ATWS Emergency Procedures
- Example ATWS Event Scenarios
- BWROG ATWS Stability Analysis
- Fuel Thermal Hydraulic Conditions

The stability requirement comes out of 10 CFR 50 Appendix A, General Design Criterion 12, requiring that the reactor has to be designed so that power oscillations which can exceed a fuel design limit are not possible or can readily be detected and suppressed (See Slide 3).

Stability and ATWS Licensing Requirements

GNF

- **Stability requirements:**
  - 10CFR50 Appendix A, GDC 12: Reactor designed so that power oscillations which can exceed a fuel design limit are not possible or can be reliably detected and suppressed
- **Stability requirements met in various ways:**
  - Exclusion regions
  - Interim manual scram requirement
  - Stability monitors
  - Automatic scram systems

The way this requirement has been addressed is that various options to meet the stability requirements have been developed. Various plants have chosen different options.

Some plants have chosen to have an exclusion region, such that it is not allowed to operate at less than e.g., 40% flow and at the power higher than, e.g., the 80% rod-line. If that region is entered, then an automatic scram is implemented for the plant. The region has been chosen because this is where it has been observed, both in calculation and in actual plant data, that an unstable situation can occur. These are the Option 1-A plants. There are various ways of monitoring to the exclusion region for the different plants, for some plants, like the option 1 -D plants, an administrative requirement to exit the exclusion region exists either by increasing the flow or by inserting control rods, if the region is entered. Finally, for other plants, a detect and suppress system has been implemented, where hardware has been installed that detects and suppresses oscillations. The hardware is able to detect both core-wide and regional oscillations and they will automatically scram on instabilities. These are the option 2 and the option 3 plants.

Stability and ATWS Licensing Requirements

GNF

- **ATWS requirements:**
  - 10CFR50.62: Each BWR must have:
  - (1) An Alternate Rod Insertion (ARI) system,
  - (2) An adequate standby liquid control system (SLCS), and
  - (3) An automatic recirculation pump trip (RPT)
- **ATWS compliance demonstrated by analysis:**
  - Evaluate limiting ATWS events
  - Do not assume ARI functions; there is no control rod insertion
  - Assume SLCS and RPT function as designed
  - Demonstrate that reactor integrity (peak RPV pressure), coolable core geometry (peak clad temperature), and containment integrity (peak suppression pool temperature) are maintained

For ATWS, the requirement is that each BWR must have an alternate rod insertion system, standby liquid control system capable of achieving hot shutdown and cold shutdown, and automatic recirculation pumps trips (See Slide 4). The limiting ATWS events are evaluated to demonstrate that the ATWS criteria is met.

In this event, it is assumed that scram fails, and that the alternate rod insertion also fails, which means that there is no control rod insertion. In the analysis it is assumed that the standby liquid control system and recirculation pumps are functional and function as designed. The single failure criterion is used. It is then demonstrated that the reactor integrity, which means the peak vessel pressure, is not exceeded, that coolable geometry is maintained, that the heat clad temperature is not exceed, and that the fuel limits are not exceed. For ATWS, there are other concerns, such as containment integrity, but these are outside the scope of this working group, and will not be covered in this presentation.

ATWS Emergency Procedures GNF

Emergency Procedure Instructions for ATWS:

- **Attempt to drive control rods**
  - Use ARI, manually insert single rods if necessary
- **Runback/trip recirculation pumps**
- **Inject boron with SLCS**
  - Use alternate method if SLCS fails
- **Bypass low water level MSIV closure signal**
  - To maintain main condenser as heat sink
- **Reduce reactor water level**
  - Reduces natural circulation flow rate and reactor power level
  - Control setpoint as low as ~2 ft below TAF (collapsed level)
- **When SLCS sufficient for hot shutdown injected, restore water level to normal range**
  - Reactor is shutdown

The emergency procedure instructions to the operators on an adverse event is that if an event without scram occurs, the operator is to attempt to use the alternate rod insertion, and to try to manually insert single rods if necessary. If that does not function, then the recirculation pumps are automatically tripped and run back to low speed, and the standby liquid boron control system is started. The low water level isolation is bypassed to maintain the main condenser as the heat sink. The primary purpose of that is to avoid containment heat-up, and minimize heat dumped into the containment.

The major action for the operator is to reduce the reactor water level. It is typically reduced down to as low as two feet below top of the active fuel. The purpose of the reduced water level is to reduce the natural circulation flow and the reactor power level. In addition, once the level is down to top of active fuel plus approximately five feet, the feedwater sparger is uncovered. When the feedwater sparger is uncovered, the feedwater is injected into a steam environment and there is almost total condensation raising the feedwater temperature close to saturation. This removes the inlet subcooling to the core, which further reduces the power, and when that happens, the oscillations are suppressed.

The water level is the maintained low until enough boron has been injected into the system to achieve hot shutdown. At that point, the water level is brought back up to normal water level, and that terminates the event.

The following slides (Slide 6-8) show a couple of typical ATWS scenarios. The MSIV closure event (See Slide 6) starts with closure of the main steam isolation valves, and the scram fails. The valves are fully closed up in about four seconds.

Example ATWS Event Scenarios GNF

MSIV Closure

• **Plant Condition**	**~Time**
– MSIV starts closing	0 sec
– SCRAM fails	0 sec
– MSIV fully closed	4 sec
– Pressure at relief set point, S/RVs open, Pool heatup begins	4 sec
– Pressure at ATWS set point, ARI fails, RPT initiated	5 sec
– Reactor isolated, No scram, S/RVs open, FW coast down/trip	0-2 min
– HPCS initiates on low water level	2 min
– Initiate SLCS, Prevent ADS, Begin level reduction	2 min
– Water level controlled at low level	5-18 min
– Boron for hot shutdown injected	18 min
– Restore water level, Hot shut down achieved	20 min

Minimal Fuel Enthalpy Increase

6

The pressure increases, the safety relief valves open, and the reactor system enters a mode of slow pressure oscillations that are controlled by the opening and closing of the safety relief valves. The reactor is isolated and there is no scram. Next, the feedwater trips, the flows coasts down, and as there is very little feedwater flow. Therefore, there will be low to no core inlet subcooling. The high pressure core spray system initiates on low water level approximately two minutes into the event. The high pressure core spray injects into a two-phase steam environment, there is close to complete condensation on the spray water, and the core inlet subcooling remains low.

The boron injection is initiated about two minutes into the transient. The operator prevents automatic depressurization and controls the water level through the HPCS injection. The water level is reduced down to top of active fuel, or top of active fuel minus two feet. The purpose of that is not only to reduce the magnitude of any possible oscillations, but also to reduce the total power level, because heat-up in containment is a concern for ATWS evaluation. Once enough boron has been injected to achieve hot shutdown, which happens roughly 20 minutes into the event, then the water level can be restored back up to normal water level. Shut down is achieved and the event is terminated.

There are some power oscillations in this events, but they are primarily associated with the closure of the safety relief valves. Each time the valves close, the system pressurize, and there is a power pulse caused by the void collapse and the associated reactivity feedback. The oscillations are relatively minor and do not exceed the fuel design limits.

There are two other types of events that lead to ATWS instability. One is the event where both recirculation pumps are tripped, and the other event is a turbine trip, both without scram.

The core flow coasts down following a trip of both recirculation pumps (See Slide 7), while the feedwater flow control is controlling the water level. The core flow coast down typically takes about up to a minute. As the steam flow is reduced because of the low power, there will be less steam extractions for the feedwater heaters and the feedwater temperature will gradually reduce. This is what causes the oscillation to start, and as it was also seen in the LaSalle event.

At LaSalle, the oscillation lasted about seven minutes into the event, at which point it reached the 118% APRM scram set point and the reactor scrammed. If that scram fails, the oscillations will continue. The operators will then start the boron injection and lower the water level. That is required per the emergency procedure guidelines. Therefore, in order to get into the very severe oscillations, one must assume that the scram fails and that the operators do not take the actions they are required to take. The start of the boron injection and the lowering of the water level will occur 7-10 minutes into the transient.

The lowering of the water level reduces the magnitude of the oscillations. Again, the operators perform the same actions. Once enough boron has been injected to achieve hot shutdown, then the water level is brought back up to the normal level and the event is terminated. This will happen 25-30 minutes into the event.

Example ATWS Event Scenarios

GNF

Both Recirculation Pumps Trip

• Plant Condition	~Time
– Both recirculation pumps trip	0 sec
– Core flow coast down, FW controlling level	0-1 min
– FW temperature reduced, Oscillations begin	2-5 min
– Oscillations reach scram set point, Scram fails	5-7 min
– Start SLCS, lower water level	7-10 min
– Oscillations reduced	10-25 min
– SLCS gives hot shutdown, oscillations terminated	25-30 min

ATWS - Stability
Intermediate Power Oscillations Possible
Boiling Transition Possible
Small cladding temperature oscillations

7

This is typical of what is referred to as ATWS instability. The power oscillations can get large, depending on how much core inlet subcooling there is. Boiling transition is possible and cladding temperature oscillations can occur, if the magnitude of the oscillations is not reduced.

The last event that can lead to oscillations is a turbine trip without scram (See Slide 8). The event starts with a turbine trip and failure to scram following the turbine trip. The event is analyzed with the bypass open following a turbine trip, as only one failure is assumed per the single failure criterion. The scram is assumed to fail, the alternate rod insertion fails, and as steam flow to the feedwater heater is lost, the core inlet subcooling increases. This will happen one to two minutes into the event. As the operators realize that the scram has failed, they initiate the boron injection and they initiate the level reduction. Large oscillations can occur as a result of large core inlet subcooling. As the operators reduce water level, the core inlet subcooling is reduced, and the magnitude of the oscillations is reduced.

When enough boron injection is achieved to get hot shutdown, in this case here, it will happen about 20 to 25 minutes into the transient, the water level is brought back up to normal level, and that terminates the event.

Example ATWS Event Scenarios

GNF

Turbine Trip

• **Plant Condition**	**~Time**
– Turbine trip	0 sec
– Bypass open, Recirculation pump trip	0-1 sec
– Scram fails, ARI fails	0-2 sec
– Reduced FW temperature, Increase inlet subcooling	1-2 min
– Start SLCS, initiate water level reduction	1-2 min
– Large oscillations develop	1-5 min
– Oscillations reduced	5-20 min
– SLCS gives hot shutdown, oscillations terminated	20-25 min

ATWS - Stability
Large Power Oscillations Possible
Boiling Transition Possible
Cladding temperature oscillations

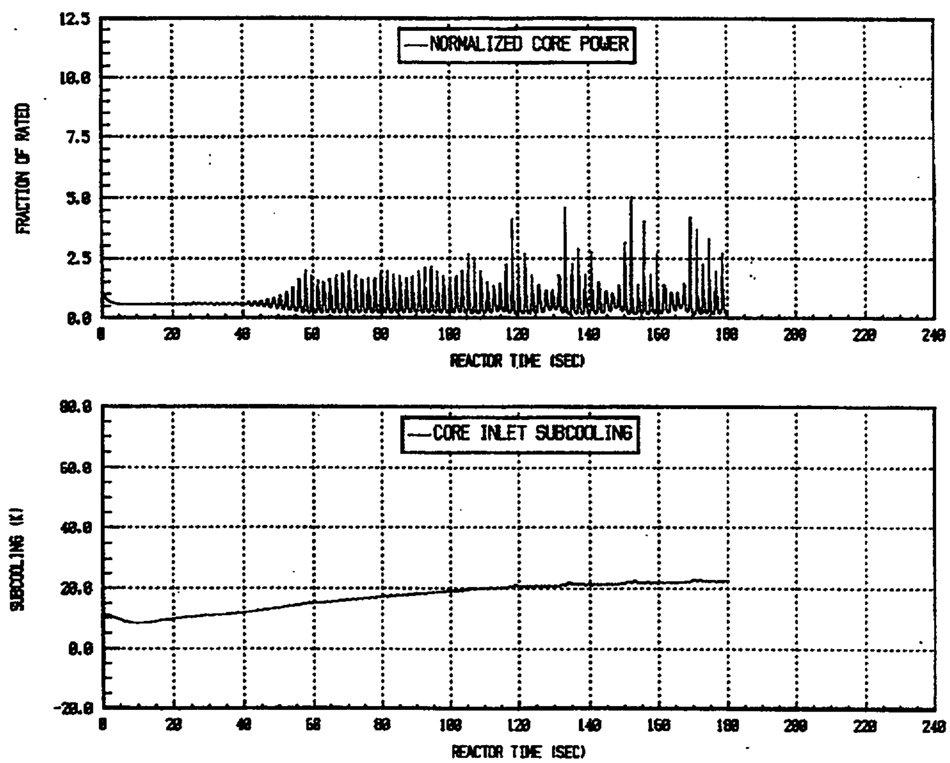
8

This scenario is very similar to the scenario following a two recirculation pump trip. Large power oscillations are possible, there can be temperature oscillations at the cladding and boiling transition on some fuel channels if the magnitude of the oscillations is not reduced.

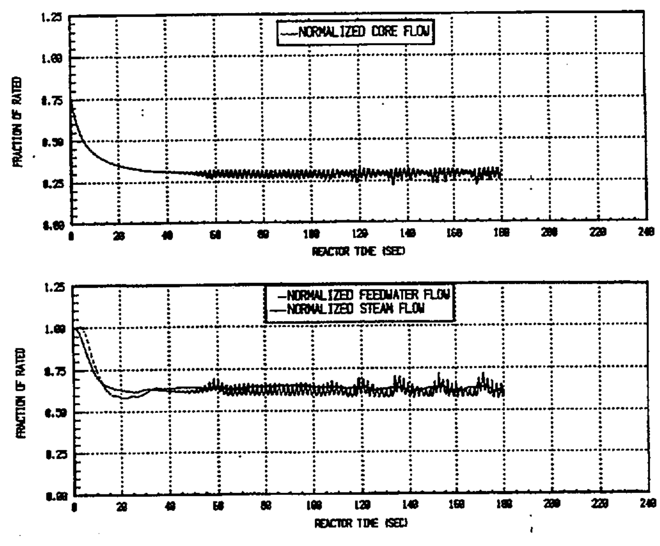
The following slides (Slide 9 and 9a-9c) show some typical examples of what an ATWS instability can look like. This was an event that was analyzed both at GE and it was also analyzed in Brookhaven following the LaSalle event. The final analyses were quite similar. This first slide shows the overall core power and core inlet subcooling. The scale on the axis for the core power is the ratio to rated core power. It starts at rated power corresponding to one and then drop down to a lower level corresponding to the natural circulation conditions. When the pumps are tripped, there is a large increase in void fraction in the core and the core power drops down to about 40 percent Then as the vapor is swept out of the core, the power comes back and settles at a power level of about 50-60% of rated. The core inlet subcooling starts around about 10 C. Initially it drops, and then as the feedwater heating is lost, the subcooling gradually increase. This is where the very large oscillations will occur. The oscillations can get very large and there can be large pulses in core average power, up to 10 times rated power and intermittently even higher pulses are possible.

BWROG ATWS Stability

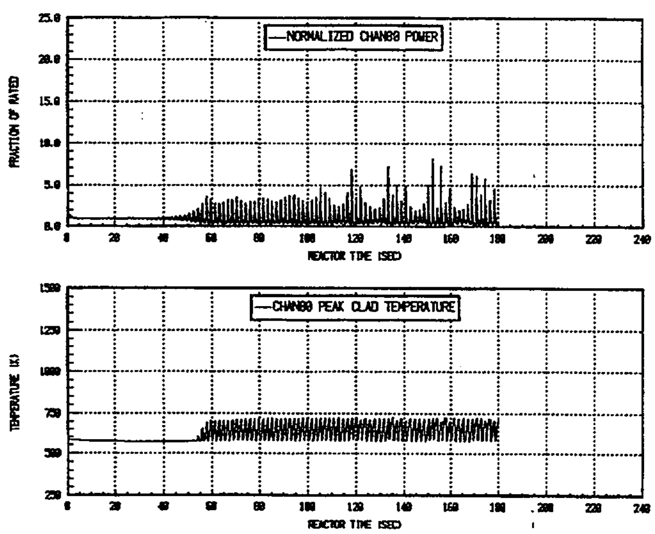
- **Typical ATWS-Stability Analysis (BWROG analysis)**
  - Core power
  - Core flow
  - Core inlet subcooling
  - High power bundle power
  - High power bundle cladding temperature
- **Analysis demonstrated**
  - For limiting conditions, power spikes can become very large
  - Hot rod dryout can occur
  - EPG actions are effective for mitigating large power oscillations



CORE POWER AND INLET SUBCOOLING
(BWR/5 RPT, CORE-WIDE OSCILLATIONS)



CORE AND VESSEL MASS FLOW RATES
(BWR/5 RPT, CORE-WIDE OSCILLATIONS)



HOT CHANNEL POWER AND PCT FOR LARGE OSCILLATIONS
(BWR/5 RPT, CORE-WIDE OSCILLATIONS)

When the oscillations are large, they become very erratic. The reason the oscillations become erratic is associated with the density wave oscillations in the channels. The density waves in the channels are controlled by the transit time from the inlet to the exit. When there is a perturbation in the inlet flow there will be perturbations in void fractions and power. In order to get an instability, there needs to be a 180 degree phase shift such that the perturbation of the pressure drop at the exit is 180 degree out of phase with the inlet pressure drop perturbation.

In the core, there are bundles with large variations of power level, and therefore the transit time from the inlet to the exit, which is given primarily by the vapor generation in the channel, will also vary from channel to channel. The transit time is not exactly the same in each channel. As large oscillations set in, they become erratic and sometimes two channels will be in phase and sometimes they will be slightly out of phase. If a large number of channels happens to be exactly in phase, the void reactivity will be in phase, and it can lead to very large power pulses. The average power, however, is relatively minor, because on average for natural circulation, the power level is in the order of 60% of rated power (See slide 10).

Fuel Thermal Hydraulic Conditions GNF

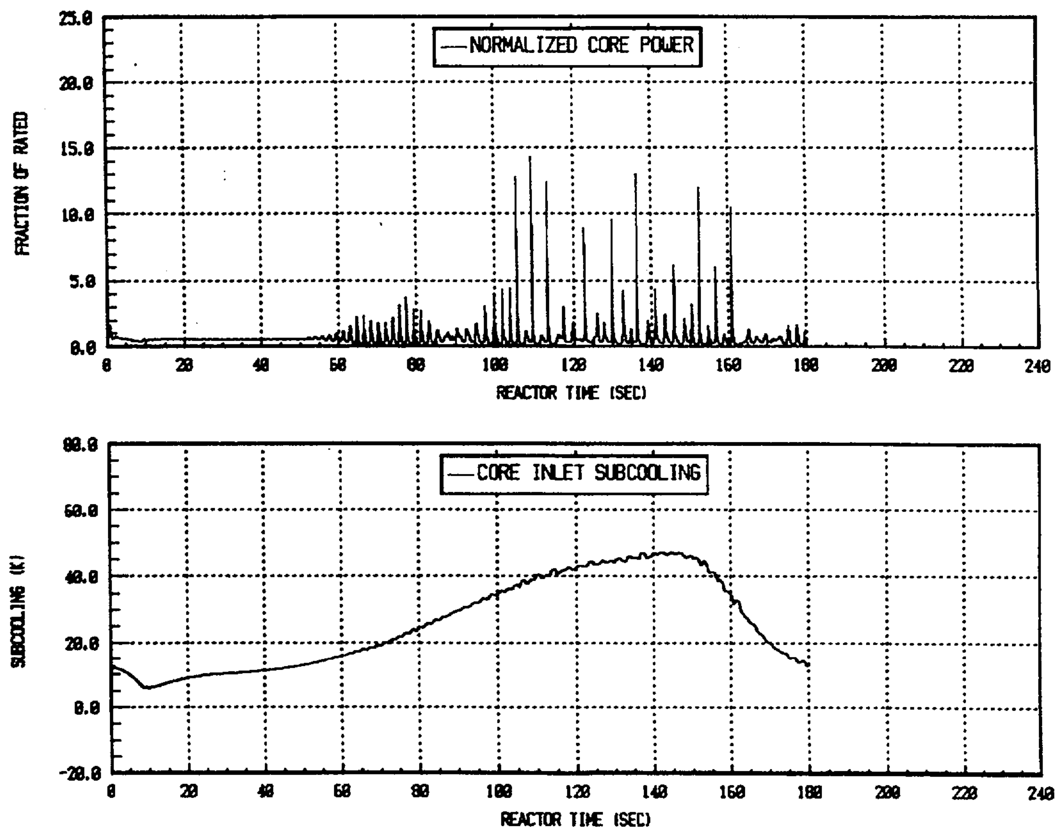
- **Two-phase water level above top of active fuel**
  - Plenty of liquid in the channel
  - Core average flow at natural circulation: ≈ 30% of rated flow
  - Average core power: 60-65% of rated power
- **Large irregular power and flow oscillations**
  - Average power oscillations: 300-500% peak power
  - Frequency: 0.5 Hz
  - Intermittent large power spikes: ≈ 1000% peak power
  - Large flow oscillations with flow reversal at inlet
  - Repeated boiling transition and rewetting for high power bundles
  - Cladding temperature increase during oscillations: 100-300K
  - If minimum film boiling temperature is exceeded, extensive cladding heatup could occur

10

The power in a two second period, which is the time period of the oscillations, can easily be calculated and corresponds to 15 to 20 calories per gram for the average oscillation. For the average peak power, those are relatively minor enthalpy generation levels. Once very large sporadic pulses exist, there can be larger enthalpy generation, and it can locally be larger for individual channels. Most of the oscillations, however, are relatively low in enthalpy.

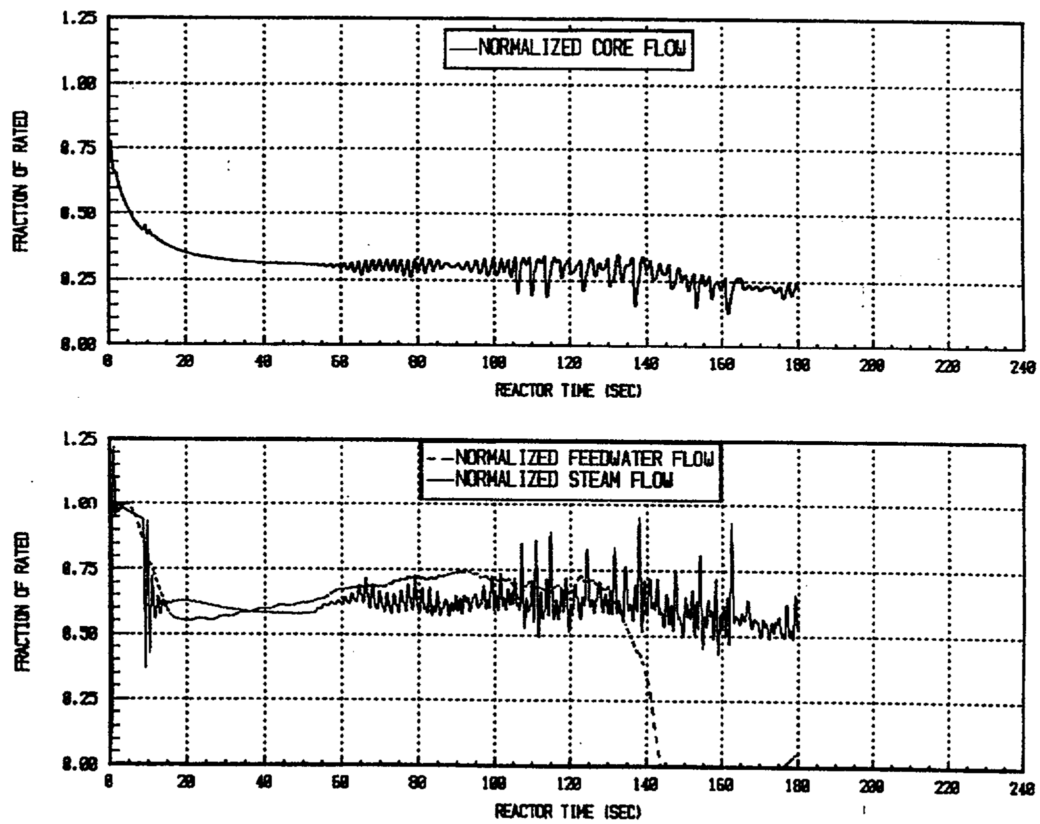
The next slides show the core flow, steam flow and the feedwater flow during the same period. There are oscillations, but they are not extremely large. Even though there are very large average power pulses up to 10 times rated power, the pulses occur on a frequency that is about two seconds. The typical fuel time constant, which is given by the heat capacity and heat transfer characteristics of the fuel, is on the order of six to seven seconds, a little less for the modern ten-by-ten fuel diameters and a little higher for the old eight-by-eight fuel. Therefore, when an oscillation occurs at a period of two seconds and the fuel time constant is about six to seven seconds, there will be a substantial damping with that power oscillation, and the oscillation in the surface heat flux will substantially smaller. This is why large oscillations in the energy deposition into the fluid are not seen, and this is also why large pressure oscillations in the system are not observed.

Slides 10a-10c are an example for a turbine trip ATWS instability. In this calculation, the emergency guidelines, where the operators cut off the feedwater flow to lower the water level, have been simulated. It happens here at about 140 seconds into the transient and leads to the drop in the core inlet subcooling and reduction in the magnitude of the oscillations.



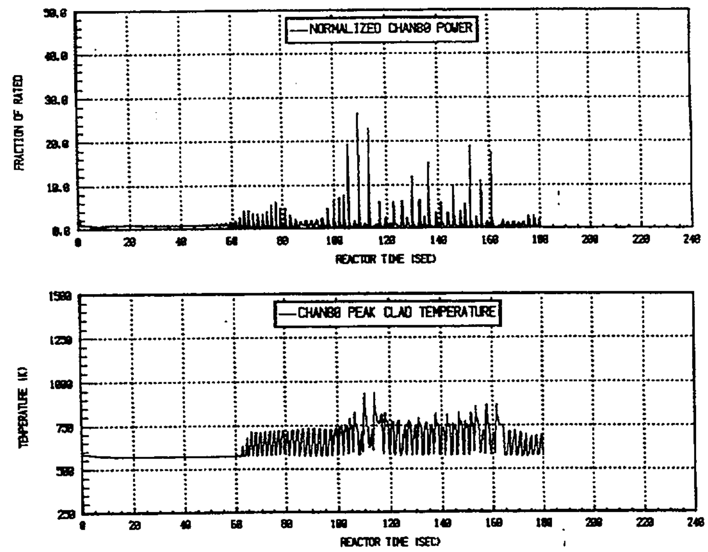
CORE POWER AND INLET SUBCOOLING
(BWR/5 TT W/ BP, CORE-WIDE, OPERATOR ACTIONS)

For this event, the hottest channel in this calculation had a radial peaking of about 1.5, which means that that fuel bundle had a power that was about 1.5 times the average bundle power. Larger power oscillations relative to its initial power are observed, whereas the power pulses for the average core went up to 10-15 times rated power, this high power channel experiences power pulses up to 20-25 and almost 30 times rated power. It is possible to locally have very large power pulses. In these calculations boiling transitions occurred for most of these pulses. However, the core is completely covered with water in this event, the downcomer water level at the top of active fuel and the two-phase level inside the core shroud is substantially higher. There is plenty of water inside the core. Therefore when a large power pulse leads to a boiling transition, the cladding quenches again, because as the power dies down, the water flows back in the channel.



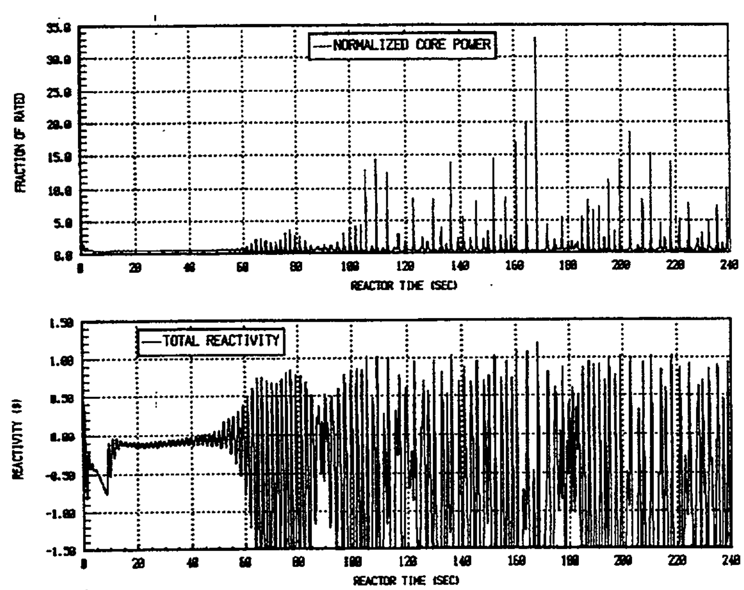
CORE AND VESSEL MASS FLOW RATES
(BWR/5 TT W/ BP, CORE-WIDE, OPERATOR ACTIONS)

The temperature rises for the oscillations are in the order of 200 to 250 degrees Celsius. The important condition is, if a temperature rise that goes beyond the minimum point on the boiling curve, because that is the point where there is no rewet. This will occur at a temperature of approximately 650 C. The channels that are operating this way are at low exposure in order to have a radial peaking factor of 1.5, typically in the order of 15-20 GWd/t. For higher exposures up to 40-50 GWd/t, it is not possible get the high radial peaking or the corresponding large power oscillations.



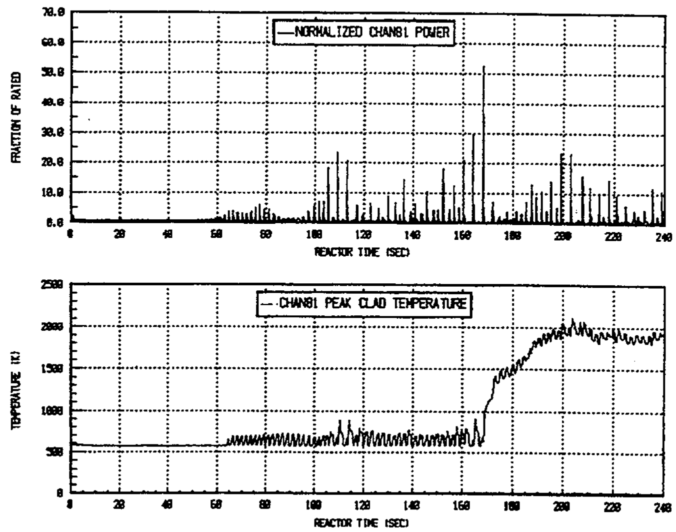
HOT CHANNEL POWER AND PCT FOR LARGE OSCILLATIONS
(BWR/5 TT W/ BP, CORE-WIDE, OPERATOR ACTIONS)

Slides 10d-10g are examples on a very conservative scenario, where it was assumed that there were no EPG actions to lower the water level and reduce the inlet subcooling. When that is done very large oscillations can occur. This event is very similar to the previous event, except that for this event the period of large irregular oscillations last much longer. By not lowering the water level, the high inlet subcooling is maintained, and eventually there will be a pulse that is huge. In this case there is a pulse at approximately 30 times rated core power. For the fuel channels the highest radial peaking was approximately 1.5. The same oscillations are observed, and eventually when the large pulse occurs, it can locally be very large.



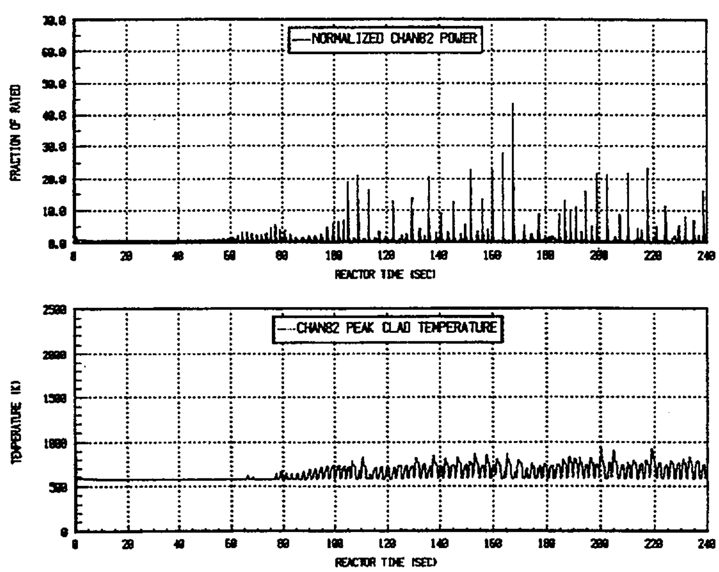
CORE POWER AND REACTIVITY
(BWR/5 TURBINE TRIP WITH BYPASS, CORE-WIDE OSCILLATIONS)

For the high power channel it went up to 60 times its initial power. If such a power pulse occurs, the temperature following the boiling transition can rise beyond the minimum film boiling temperature, there will be no rewet, and very large temperatures can result.

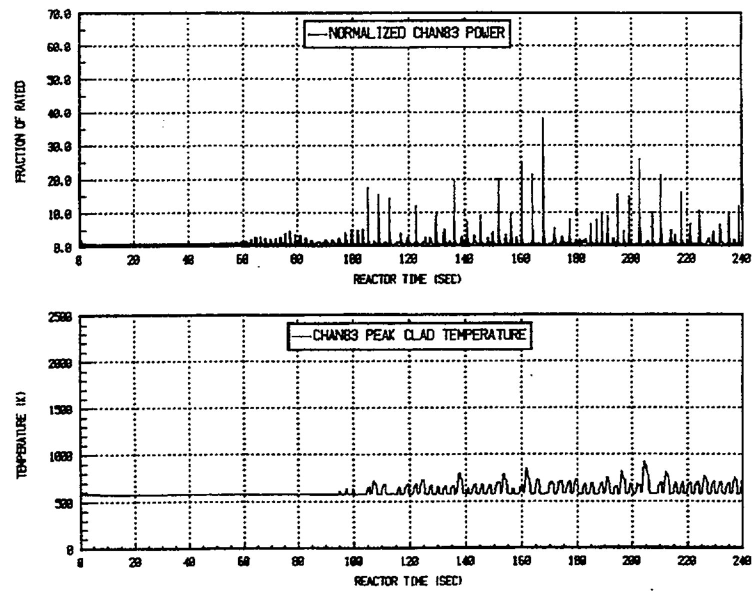


AVERAGE (RPF=1.45) CHANNEL POWER AND PCT FOR LARGE OSC.
(BWR/5 TURBINE TRIP WITH BYPASS, CORE-WIDE OSCILLATIONS)

The next two slides show the transient for channels with lower radial peaking and it is seen that the severe temperature excursion only occurs for the channels with the highest radial peaking, in this case greater than 1.4.

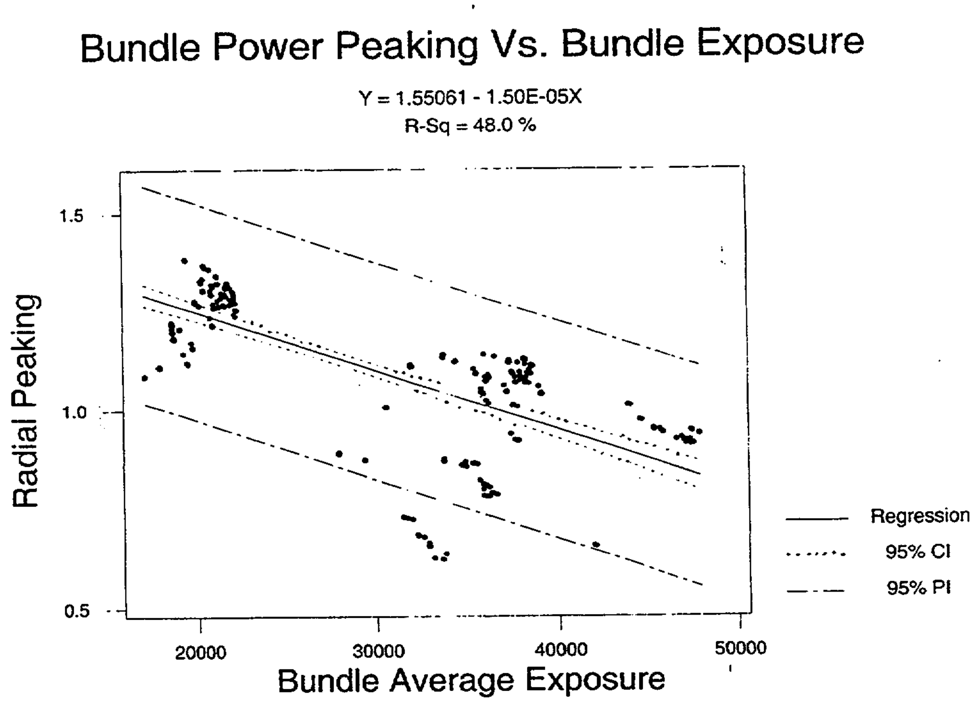


AVERAGE (RPF=1.38) CHANNEL POWER AND PCT FOR LARGE OSC.
(BWR/5 TURBINE TRIP WITH BYPASS, CORE-WIDE OSCILLATIONS)



AVERAGE (RPF=1.19) CHANNEL POWER AND PCT FOR LARGE OSC.
(BWR/5 TURBINE TRIP WITH BYPASS, CORE-WIDE OSCILLATIONS)

Slide 10h is a plot of radial peaking versus exposure for a typical BWR core, and it is seen that the high radial peaking only occurs for low fuel exposures. For high exposures, there is not sufficient reactivity left in the fuel to get radial peakings above the average power. Therefore large temperature excursions will not occur for high exposure fuel during an ATWS instability.



To summarize the presentation (See Slide 11), for limited conditions, power pulses can become very large. To get into those limiting conditions, very conservative assumptions must be made, all scram system and alternate rod injection systems must fail, and one must assume that the operators do follow the emergency procedure guidelines. This makes it an N-2 type event. For the other situation, where the operator follows the emergency procedure guidelines, the large erratic oscillations will not occur.

Impact of Burnup on ATWS Stability GNF

- **Large Power and Temperature Oscillations Occurs for Bundles with High Radial Peaking, 1.5+**
  - High radial peaking occurs for peak reactivity
  - Late in first cycle when Gadolinium is burned off
  - 15-20 GWd/t
- **High Exposure Fuel Has Low Reactivity and Low Radial Peaking**
  - Low power and small power oscillations
  - Larger CPR margin and small temperature oscillations
  - Small enthalpy deposition due to oscillations

Large Oscillation and Enthalpy Deposition Not Expected for High Burnup Bundles
Similar to Conclusion from Rod Drop Analysis

11

There may be some oscillations for a short period of time, but they will not last for very long. The magnitude of the enthalpy addition for the high exposure bundles would be very low, in the range of less than 10-20 calories per gram. Dryout can occur on the hottest bundles, but those are the bundles at low exposure, typically 15-20 GWd/t.

Finally, the calculations show that the emergency procedure guidelines are very effective in mitigating the large oscillations.

To summarize the conditions in the core for ATWS instability, the water level is at the top of the active fuel, the two phase water level is substantially higher, and there is plenty of fluid available to cool the fuel bundles. The core average flows at natural circulation is approximately 30% of rated flow, and the average core power is typically 60-65% of rated power for these events. There can be large irregular power oscillations.

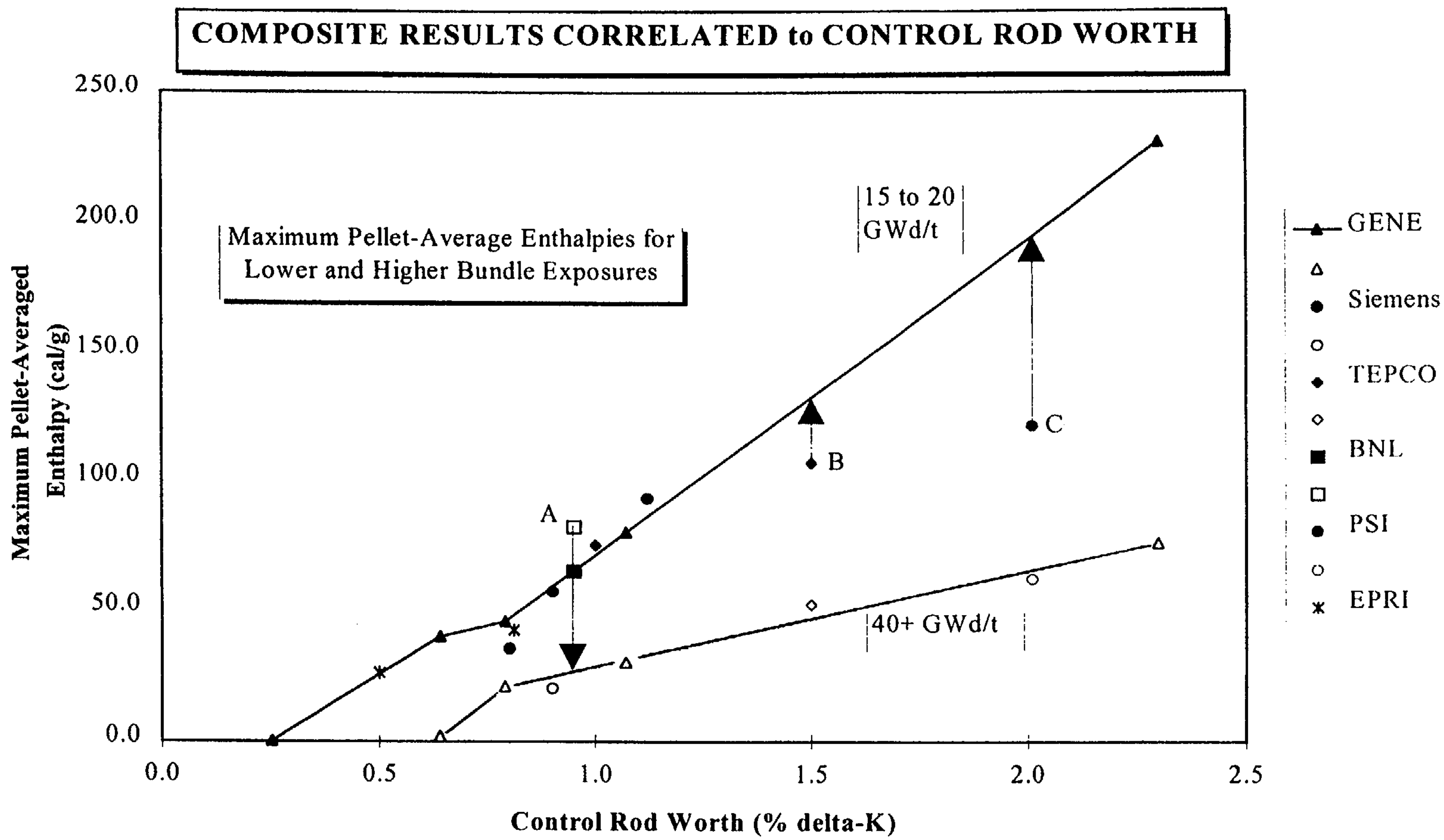
The average power oscillations are in the 300-500% percent range. Intermittently, there can be very large pulses, over 1000% of rated power. Large flow oscillations can exist for these conditions. When a large power pulse occurs, the vapor generation, particularly from instantaneous direct moderator heating expels the water out of the bundle. Flow reversals at the inlet will occur and that is what drives the boiling transition, while the heat flux oscillations are relatively minor because of the larger fuel time constant.

Repeated boiling transition and rewet is possible, and therefore the phenomena associated with boiling transition and rewet are very important. During oscillations, the cladding temperature oscillations may range from 100° to 300° K for the oscillations. The 300° K are for the fresh bundles, the 100° K or less is for the high burnup bundles. If minimum fuel boiling is exceeded, which is around 900° K, then extensive fuel heat-up can occur. This heat-up, however, will not happen for high burnup bundles, but is only likely for fresh bundles.

Therefore, in conclusion, large power and temperature oscillations can occur for bundles with high radial peaking, in the order of 1.5. For high exposure fuel that has low reactivity and low radial peaking, there will only be small power oscillations. Because of the low power and the higher flow in these channels the CPR margin will be large, and as a result, boiling transition is not likely and the corresponding temperature oscillations will be small. The enthalpy deposition due to the oscillations is relatively small, and large oscillations and enthalpy deposition is not expected for high burnup fuel bundles. This conclusion is similar to the conclusion that was obtained when the impact of fuel exposure was analyzed for the rod drop event.

Enthalpy Increase for BWR RDA

G-82



Pellet-Averaged Enthalpy Depends on Control Rod Worth
Pellet-Averaged Enthalpy Decreases at Higher Exposures

NRC FORM 335
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10. SUPPLEMENTARY NOTES

H. Scott, NRC Project Manager

11. ABSTRACT *(200 words or less)*

In the United States, two types of regulatory criteria have been used in safety analysis to address reactivity accidents. One criterion is a limit of 280 calorie per gram of fuel on peak fuel-rod enthalpy. The other criterion consists of several threshold values that are used to indicate cladding failure. The U.S. Nuclear Regulatory Commission (NRC) is performing research with respect to high burnup fuel to acquire and develop the requisite understanding of the performance of high burnup fuel under accident conditions. The NRC is also preparing to develop a new criterion to replace the current enthalpy limit. To support these efforts, the NRC has commissioned the formation of a Phenomena Identification and Ranking Table (PIRT) panel to identify and rank the phenomena occurring during selected transient and accident scenarios. Membership of the PIRT panel has been drawn from the US and international scientific community and many of its sixteen members are actively involved in experimental and analytical work related to the behavior of high burnup fuel under accident conditions. Because the PIRT identifies and ranks phenomena for importance, currently existing experimental data, planned experiments, computational tools (codes), and code-calculated results can be screened to determine applicability and adequacy using the PIRT results. The scenario described in this report is flow instability driven power oscillations arising during a postulated anticipated transient without scram in a Boiling Water Reactor containing high burnup fuel. Although a specific plant and fuel have been selected, the panel was charged with the responsibility of extending the applicability of the PIRT to cover other fuel, cladding, and reactor types and fuel burnups to 75 GWd/t.

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