DETERMINATION OF IN-VESSEL RETENTION UNDER MOLTEN CORIUM POOL ATTACK

D. Home and M. Chai
Risk Management Directorate, Amec Foster Wheeler
700 University Avenue, Toronto, ON M5G 1E6, CANADA
deep.home@amecfw.com; michael.chai@amecfw.com

ABSTRACT

This paper discusses the development of Decomposition Event Tree (DET) logic to capture and address the physical challenges and the uncertainties associated with the molten corium pool attack on the Reactor Pressure Vessel (RPV) lower head under Internal Reactor Vessel Cooling (IRVC)/ External Reactor Vessel Cooling (ERVC) conditions. This study assumes that core debris has relocated to the bottom of the RPV lower head i.e., mitigating strategies to prevent core disassembly and relocation to the RPV lower head have been unsuccessful. The technical issues that are covered in this paper include the morphology of corium pool formation and its transient characteristics, phase separation of the corium melt, natural convection heat transfer inside the corium pool, solid crust formation, molten chemical attack of the vessel wall due to Fe-Zr eutectic reaction, corrosive action of molten corium on the vessel wall, and creep deformation of the RPV lower head due to thermal and mechanical loads. The paper proposes guidance on performing conservative and bounding analytical calculations with consideration of the uncertainties to address the aforementioned RPV challenges from the molten corium pool so as to develop a robust argument for In-Vessel Retention (IVR) under ERVC conditions. As described in this paper, a systematic approach to determine the likelihood of IVR should be documented in a site Level 2 PRA/PSA study.

KEYWORDS
In-Vessel Retention (IVR), Internal Reactor Vessel Cooling (IRVC), External Reactor Vessel Cooling (ERVC), Decomposition Event Tree (DET), Level 2 PRA/PSA

1. INTRODUCTION

This paper identifies the physical issues associated with the molten corium pool behavior and discusses the ensuing thermo-chemical-mechanical challenges that the molten corium pool poses to the integrity of the Reactor Pressure Vessel (RPV) lower head under External Reactor Vessel Cooling (ERVC). The physical challenges to the RPV structural integrity, ensuing from severe accident scenario are captured through a Decomposition Event Tree (DET). The generic DET is developed to identify the top physical events, in a sequential time frame, when core debris is present at the bottom of the vessel. Guidelines for performing conservative analytical calculations are presented for the scenario when molten corium pool resides in the RPV lower head under ERVC conditions, these guidelines must be considered when claiming IVR from a Level 2 PSA perspective. The analytical calculations in the Level 2 PSA framework are generally performed based on a steady state bounding approach (taking the uncertainty bands into account), these calculations serve to complement the assessments from the severe accident analysis codes like MAAP, MELCOR etc. Further, the calculations undertaken should be aimed at providing the probability for different vessel failure modes under ERVC condition.

2. BACKGROUND

The issues pertaining to In-Vessel Retention (IVR) have gained attention following the Fukushima Daiichi incident and the nuclear industry's investment in FLEX. Retention of the core debris within the Reactor Pressure Vessel (RPV) lower head in a beyond design-basis accident will limit the consequences of a severe accident. Conversely, RPV failure as a result of the thermo-mechanical loads applied by the core debris melt may result in significant consequences with regards to ex-plant radiological release and flammable gas generation that can potentially fail containment. The success of Severe Accident Management (SAM) strategies, such as Internal Reactor Vessel Cooling (IRVC)/External Reactor Vessel Cooling (ERVC) to prevent vessel failure and retain the core debris inside the RPV, are primarily dependent on the molten pool dynamics and thermo-chemical-mechanical demands on the RPV lower head wall. Stated another way, the corium (i.e., molten core) behavior in the lower plenum of the reactor vessel is very important as it affects the thermal and mechanical loads that the RPV wall is subjected to, and this thermo-physical behavior ultimately governs the accident development. Further, it needs to be acknowledged that there are significant uncertainties associated with the multi-component corium mixture that affects the corium behavior and its interaction with the RPV wall.

The in-vessel considerations of a severe accident can be categorized into two time frames viz., (a) early time fame and (b) median-to-late time frame. In the early time frame the core disassembles, partially melts and relocates to the lower plenum (bottom of the reactor vessel), where it interacts with the residual water to form a complex particulate debris bed characterized by interstitial porosity. In the median-to-late time frame of the accident progression the particulate debris bed re-melts due to the decay heat of the fission products and forms a multi-component molten mixture (corium pool) that is characterized by internal natural convection currents which transfer the decay heat in various directions. The SAM strategies of IRVC and ERVC to retain the core debris inside the vessel is related to the aforementioned time frames.

The SAM strategy of ERVC to retain the core debris inside the vessel is employed by submerging the vessel completely or partially up to the lower head in a water pool. The ERVC strategy comes into effect during the late time frame of the in-vessel accident progression when a coherent molten corium pool is formed in the lower plenum. The ERVC method to support IVR has been evaluated in detail with regards to its effectiveness to maintain the RPV integrity for medium powered AP600 reactors [1], the concept has been further extended for AP1000 reactor plants [2]. The Finnish Loviisa VVER-440 nuclear power reactor has already implemented the SAMG strategy of flooding the reactor cavity to arrest the progression of a core melt down scenario and retain the core debris in-vessel [3]. For the Loviisa plant, several design modifications[1] were incorporated to ensure efficient natural circulation of coolant outside the RPV [4]. Preliminary calculations based on thermo-mechanical loads for the medium and large power VVER-600 and VVER-1300 reactors have shown that maintaining an effective ex-vessel cooling provides positive arguments in favor of application of the IVR corium concept [2]. Ex-vessel cooling as an accident management strategy to substantiate IVR has initially been proven for APR1400 [5] however, further assessments are required to consolidate the facts. CANDU reactors by design have water surrounding the entire Calandria vessel (CV i.e. equivalent of RPV) during normal operation, depending on the plant design the water is held in either a cylindrical vessel (known as the Shield Tank) or in a concrete reactor vault that surrounds the CV. Thus, ex-vessel cooling by design is always available for the CANDU reactors even under severe accident scenarios. Emergency Mitigating Equipment (EME) actions are in place at the CANDU stations to supplement the water under boil-off conditions for severe accident

[1] Some of the design modifications included: sufficient minimum gap between the outer wall of the lower head and thermal insulation to streamline the flow and prevent steam choking, uninterrupted escape of steam-water mixture by installing passively operated valves into the thermal insulation surrounding the RPV outer wall, installing screening equipment in the reactor cavity to trap particles etc.

scenarios ensuring that water contact with the CV is always maintained to remove the decay heat. Small Modular Reactors (SMRs) like the one designed by NuScale Power have passive safety systems in place such that under a severe accident core melt down scenario the containment vessel (which surrounds the RPV) is flooded with water. Further, the containment vessel by design is itself submerged in a larger water pool held in a steel lined concrete structure such that overall IVR argument is favorable for the NuScale designed SMR.

It has to be recognized that proving the concept of IVR for one reactor type does not necessarily lead to an easy and straightforward extension of the concept to other reactor types. Two central factors that affect the IVR through ERVC are: a) the thermal power of the reactor, and b) the intrinsic design of the reactor. Thermal power dictates the thermal load that the vessel wall is subjected to for a core melt down scenario and, in general, the requirements to cool the vessel need to be more robust for higher power reactors. Intrinsic design covers the areas of core configuration, vessel geometry and material, and design of the reactor cavity that have significant effect on the physical loads that the vessel wall is subjected to and the ex-vessel cooling management.

Work performed in Reference [1] used the Risk-Oriented Accident Analysis Methodology (ROAAM) to clearly lay out in detail the design related and physical issues that need to be given due consideration to address the suitability of ERVC to maintain the vessel structural integrity and claim IVR[2]. Of central importance to the ERVC problem (it is acknowledged that there are other key issues as well) is the margin to Critical Heat Flux (CHF) that sets the limit for the heat removal rate from the corium debris pool [1]. Effective ex-vessel cooling can be provided by water surrounding the outer wall of the vessel and debris retention can be ensured as long as the CHF is not exceeded. Further, if CHF is surpassed i.e. the boiling regime transitions from the effective nucleate boiling mode to the vapor filled film boiling mode then boiling crisis follows, leading to the structural integrity of the vessel wall being compromised due to the imposed thermal loads from the molten corium pool. Experimental programs at the ULPU [1], [6] and SULTAN [7] facility have shed light into the details of boiling heat transfer and CHF for prototypic reactor vessel that is cooled on the outer surface by water. The ULPU facility has provided insights into external flow tailoring arrangement (to create streamlined and organized flow of the coolant) whereby higher values of CHF are obtainable, as high as ~1.8 MW/m^2 at the top azimuthal angles (60^o) of the vessel [8], which is beneficial for medium to high power reactors when claiming IVR.

The SAM strategy of IRVC is related to the early time frame of the in-vessel accident progression whereby the vessel is flooded with coolant water to arrest the progression of severe accident and render the core debris in a coolable configuration (prevent it from reaching the configuration of a coherent molten pool). The hot and dry particle debris bed is flooded with water either from top (top-flooding) or from bottom (bottom-flooding), with a possibility of lateral flooding of the debris bed as well. Experimental research programs at the test facilities of DEBRIS, PRELUDE, POMECO [9-10] etc. have looked in detail regarding the physics issues that surround the in-vessel cooling methodology. The two primary focal points of these test programs are the characterization of the frictional pressure drop, and the dry out heat flux that sets the limit on the coolability of the porous debris bed, which in turn depends on the Counter Current Flow Limit (CCFL). The tests have shown that bottom-flooding is the most effective in cooling the debris bed in-vessel as the CCFL is not applicable in this flooding mode thereby rendering larger dry out heat fluxes due to enhanced flow penetration into the debris bed (vapor and liquid flow is in the same direction). The physical issues under IRVC need to be studied further so as to determine an effective design for the downcomer which is used to channel the coolant flow into the particle debris bed. It is recognized that physical issues related to ERVC have been studied to a much greater detail and extent as compared to the issues that surround the IRVC methodology.

---

[2] The issue was addressed for the AP600 reactor design.

Severe Accident Research Priority (SARP) work was updated under the SARNET2 project FP7 by evaluating the developments in the experimental research programs and taking into account outstanding safety issues, as illustrated by Level 2 PSA studies [11]. Some of the areas for further study to reduce uncertainties with regard to IVR are [11]:

- Corium behavior in the lower head: Of particular importance are 2 issues (a) influence of Control Rod Guide Tubes (CRGTs) on the corium melt behavior in the lower plenum for BWRs, and (b) transient nature of the molten pool with regards to the 2-layer and 3-layer configurations and the evolution of the thickness of the layers.
- RPV integrity by ex-vessel cooling: Database of CHF needs to be further improved for the external cooling conditions so as to evaluate and design accident management strategies of ERVC for IVR.
- RPV failure mode: Model predictions concerning the location and failure mode of the RPV need to be improved particularly for the failure of the BWR penetrations. This in turn affects the corium discharge into containment.

The following section discusses the characteristics of the corium pool that need to be considered when determining whether IVR of the corium with ERVC conditions is achievable.

3. CORIUM POOL DYNAMICS

In a severe core damage accident scenario wherein the fuel channel assemblies and in-core support structures and instruments ultimately fail and relocate to the bottom of the RPV to form a terminal core debris bed, a complex multi-component mixture ensues primarily comprising of elements U-Zr-O-Fe-C-B. The composition of the elements in this mixture varies across the different reactor core types and the degree of oxidation determines the distribution of the elements in this heterogeneous terminal core debris bed. This mixture undergoes various stages of phase transformation as the residual water inside the RPV is boiled off due to the decay heat. The different stages range from an initial particulate debris bed to a coherent molten pool formation, the molten pool is associated with a solidified crust boundary (assuming that ERVC is present) that separates the melt from the RPV inner wall. The focus of the following sections is to describe the characteristics of this molten pool during the late stages of in-vessel accident progression so that the reader can appreciate the physics that must be considered in the Level 2 PSA DET.

3.1. Morphology and Transient Characteristics of Corium Pool

The geometry of the core debris at the bottom of the RPV will evolve following core collapse. The initial debris geometry will be largely a porous debris bed. The debris bed will be comprised of disassembled fuel channels and in-core support structures, instruments, etc. In the absence of heat removal via internal cooling of the RPV, heatup of the debris will lead to the generation of molten material. Should interaction between the molten debris and the RPV wall not lead to localized failures of the vessel during this melting transient, large masses of molten material will be formed in the RPV. The transient development of the various stages of the terminal debris configuration is very complicated marked by different physico-chemical interactions that cannot be determined a priori. Further, it is not necessary to have a complete specification of the intermediate configurations. The precise understanding of the melting and relocation of material from a largely porous debris bed into a largely coherent molten pool is not necessary for the purpose of assessing the RPV integrity. The primary concern to the RPV integrity following the molten corium pool formation is the magnitude of through-wall heat fluxes. In order to assess the magnitudes of these heat fluxes, it is most useful to present a set of stylized configurations of core debris that could arise. Such stylized configurations provide the ability to bound the magnitudes of through-wall heat fluxes to which the RPV assembly could be subjected to in a severe accident. This provides a manageable paradigm in which to assess the potential for integrity of the RPV under severe accident loads as well as establishing the types of failure modes that are most likely to occur. These stylized configurations represent the steady state (in real sense they are quasi-steady) of the molten corium pool after the debris melts and are confirmed through experimental tests on prototypic reactor materials and geometry.

The Light Water Reactor (LWR) experimental programs such as RASPLAV [12] and MASCA [13] have shed important light on the eventual molten corium pool configuration. In general, the molten corium pool consists of metallic materials such as stainless steel and Zircaloy, and ceramic (oxidic) components such as UO_2 and ZrO_2. The corium pool can be stratified into a 2-layer configuration or a 3-layer configuration as shown in Figure 1 and Figure 2 respectively, which are quasi-steady from the fluid flow and heat transfer perspective. In the 2-layer configuration, a metallic layer resides on top of a molten oxidic pool, with the metallic layer causing the focussing of the heat flux onto a small area that could potentially surpass the local CHF and thus, the vessel integrity could be challenged due to an increased thermal load. The oxidic molten pool transmits energy to the lower boundary, which gets distributed to the other surfaces of the metallic layer due to buoyancy driven convection currents. The upper boundary of the metallic layer radiates its energy to the vessel surfaces above it. The lateral boundary transmits its energy directly to the side walls of the RPV for which the heat flux focusing occurs. Heat flux augmentation occurs by more than a factor of 2 for metal layer heights less than 20 cm thick [14]. The heat transfer in the metal layer is governed by the Rayleigh-Benard convection flow patterns. The heat transfer across the different metal layer surfaces (lower/upper and lateral) are estimated using the natural convection based bulk correlations:

$$Nu_{l/u} = 0.069 Pr^{0.074} Ra^{1/3} \tag{1}$$

$$Nu_{lat} = \frac{0.15}{[1+(0.492/Pr)^{9/16}]^{16/27}} Ra^{1/3} \tag{2}$$

The above correlations are based on a horizontal fluid heated from below and cooled from the top [3]. Recently conducted tests at the LIVE-L6 facility have studied the layering effect with emphasis on the transient behavior (heat flux distribution along the RPV wall, melt temperature evolution, crust growth) using non-eutectic KNO_3-$NaNO_3$ melt solution [15]. The tests have confirmed the heat flux augmentation (at the periphery of layer separation), it will be beneficial to see the behavior of the heat transfer correlations in the separated top layer from these tests.

The 2-layer configuration depends on the following factors: a) Ratio of Fe/corium i.e. the amount of stainless steel mass in the core and, b) the amount of carbon or boron in the core composition. Larger values of the aforementioned factors aid in the formation of the stratified metallic layer. The 2-layer configuration has been observed through the experimental programs for the LWR cores however, this formation is not relevant and is highly unlikely for the CANDU design as per the observations from the MATICAN tests conducted recently at the RASPLAV facility for the CANDU core [16].

In the 3-layer configuration, a heavy metallic layer settles below the molten oxidic pool with a thin metal layer on top of the oxidic pool. The heavy metallic layer at the bottom is a mixture of U-Zr-Fe, which forms because uranium gets reduced from the oxidic pool by the unoxidized Zr thereby, leading to an increased density of the metallic elements and thus, settlement of the metallic elements at the bottom of the vessel. The buoyancy driven forces in this case are not able to overcome the gravitational force caused by the density difference that leads to the inverse stratification. The formation of the heavy metallic layer depends on the following factors [17]: a) Degree of Zr oxidation, b) Ratio of U/Zr, c) Stainless steel mass and, d) UO_2 mass. The 3-layer configuration has the limiting effect of decreasing the thickness of the stratified top metal layer which could be detrimental to the RPV structural integrity due to a further enhancement of the focussing effect. Further, the concentration of the metallic elements at the bottom leads to a situation where the decay heat generated inside the metallic layer gets focused at the bottom of the vessel where the margin to CHF is at its minimum. Thus, a 3-layer pool configuration is the most limiting scenario for estimating the bounding heat fluxes. It is recognized that there are no detailed analysis on the natural convection heat transfer process in the special geometric arrangement of a bottom metallic layer with a small volumetric heat generation rate, experimental tests are required to ascertain the nature of heat transfer correlations for such a bottom metallic layer inclusion.

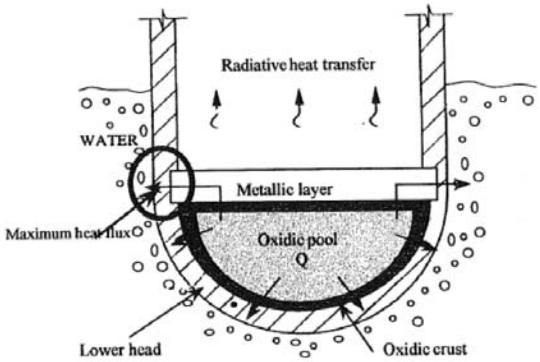

Figure 1: Two-layer corium pool configuration [18]

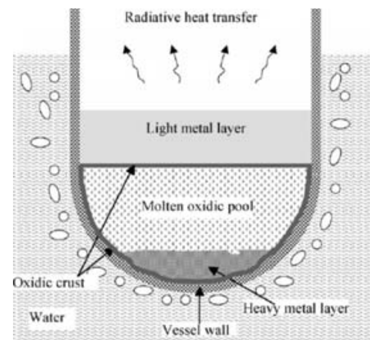

Figure 2: Three-layer corium pool configuration [17]

3.2. Molten Pool Heat Transfer

Prior to the formation of a coherent molten pool, energy transport is largely conductive and to a lesser extent radiative. Combined with the highly oxidic nature of the debris, very large temperature gradients would be required to transport the decay heat to the water surrounding the RPV. This would lead to debris temperatures well past melting. Thus, the energy transport is conduction-limited and the decay heat largely goes into the stored energy of the materials. Melting will thus occur and the continued volumetric heat generation will drive buoyancy-induced natural convection currents toward the upper surface of the debris bed. As the upward plumes of hot debris are cooled via radiative heat transfer at the upper surface of the debris bed, the increased density will lead to downward flows of molten debris along the RPV lower head downward-facing surface. The basic features of the flow regime in the molten convective pool is thought to comprise of steep boundary layer regions adjacent to the isothermal boundary, a well-mixed upper pool volume, and a broadly stratified lower pool portion [1]. The upper mixed region is created by the cold plumes detaching continuously from the upper boundary layer, and the lower stratified region is the result of the down-flowing boundary layers along the curved boundary and the induced bulk motion. The formation of a coherent molten pool will lead to the circulation of molten material to the boundaries of the pool. For debris confined within the lower head of the RPV these boundaries correspond to:

- The curved, downward-facing surfaces of the RPV lower head shell in contact with the core debris. The energy transported across this boundary will be conducted through the wall of the lower head shell and rejected into the fluid environment surrounding the outer wall of the lower head.
- The upward-facing surface of the debris pool. The energy transported through this boundary will be radiated to the calandria assembly surfaces having a direct line-of-sight to the debris bed's upward-facing surface.
- For CANDU reactors only: There are vertically-oriented walls of the Calandria assembly that are in direct contact with the core debris bed. These surfaces are the vertically-oriented portion of the notch (annular plate) and the calandria tube sheets at either end of the reactor core. Heat transported through this boundary is conducted through these portions of the Calandria assembly and rejected into the fluid environment of either the Shield Tank (notch region) or the end shield.

Due to the different orientation of each fluid boundary with respect to the convective flows in the molten pool, distinct heat fluxes through these boundaries will occur. The natural convection heat transfer associated with the molten pool is inherently transient in nature. Within a simplified analysis relevant for the purposes of a Level 2 PSA, the use of heat flux correlations determined from experiments judged to be representative is acceptable. It is however, recognized that these correlations are based on the bulk convective flow and the effect of localized convective flow pattern distribution in the molten pool is neglected. The convection flow in the molten corium pool in the lower plenum is in the turbulent regime with Rayleigh numbers of the order of ~10^{15}-10^{16} [3]. The Rayleigh number is usually stated as:

$$Ra = \frac{g\beta q''' H^5}{\nu\alpha k} \tag{3}$$

where the thermo-physical transport properties in Eq. (3) are depicted in Table 1. The additional symbols are defined as: g is acceleration due to gravity [m/s^2], $q'''$is volumetric heat generation rate [W/m^3]; $H$is height of the debris [m] and α is the thermal diffusivity [m^2/s]. The Rayleigh numbers are presented in Figure 3, with the upper limit being 10^{15}.

Table 1: Transport Parameters for Stylized Debris Beds

Transport Parameter	Symbol	Nominal Zr Oxidation Case [19]	Completely Oxidized Zr Case [19]	Zr-Migration Case [1]
Specific Heat [J/kg-K]	c	530	590	510
Thermal Conductivity [W/m-K]	k	20	4.6	4.7
Kinematic Viscosity [m^2/s]	ν	9×10^{-7}	8.5×10^{-7}	6.33×10^{-7}
Density [kg/m^3]	ρ	7560	7300	8450
Volumetric Expansion Coefficient [1/K]	β	8×10^{-5}	9.3×10^{-5}	1.05×10^{-4}

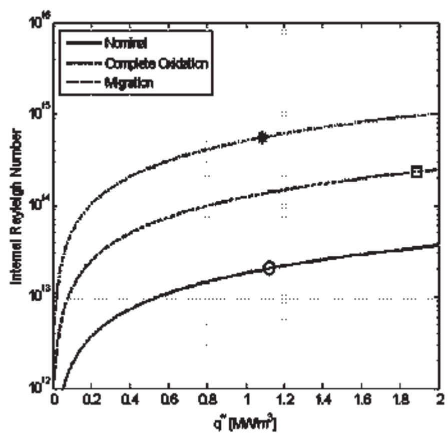

Figure 3: Variation of Rayleigh number with volumetric heat generation rate

Several experimental programs including Mini-ACOPO and ACOPO [1], SIMECO [8], COPO [8] etc., have shed important light into the convective heat transfer behavior for volumetrically heated molten pools in prototypic reactor vessels (with different linear scales), it has been generally found that about 2/3rd of the generated heat is transferred in the upward direction. The Rayleigh numbers are found to be high enough to ensure turbulent motions exist and that majority of the heat is convected upwards. The average downward and upward heat transfer coefficients (representing the bulk flow motion) for a steady state type of condition, from the ACOPO facility (hemispherical vessel) were found to be [20]:

$$Nu_d = 0.1857 Ra^{0.2304} \left(\frac{H}{R}\right)^{0.25} \tag{4}$$

$$Nu_u = 2.4415 Ra^{0.1722} \tag{5}$$

The downward heat transfer on the RPV lower head wall has a dependence on the azimuth position. In the severe accident safety analysis code MAAP, the aforementioned correlations take one of the following forms (based on Jahn and Reineke correlation):

$$Nu_d = 0.54 Ra^{0.18} \left(\frac{H}{R}\right)^{0.26} \tag{6}$$

$$Nu_u = 0.36Ra^{0.23} \tag{7}$$

The above correlations have the same functional dependency however, they differ in the coefficients and exponents. The correlations as stated above are generally used in the Level 2 PSA studies in the steady state bounding approach. Recent tests at the LIVE facility have used a large-scale 3D geometry to study the phenomena resulting from the core melting with emphasis on the transient behavior [21]. The central finding from this work was that the initial debris melting process does not influence considerably the steady state character of the melt pool i.e. the final steady state (or more precisely quasi-steady) configuration of the melt pool configuration and the associated heat transfer can be assumed to be bounding. This conclusion bears significance for performing steady state type bounding analysis using bulk type correlations for Level 2 PSA to determine IVR when ERVC is credited.

3.3. Solid Crust Formation and Gap Heat Transfer

When sufficient cooling is present on the outer surface of the RPV (there is no boiling crisis), there is formation of corium crust on the inner wall of the vessel which prevents any direct contact of molten corium with the vessel wall. If this configuration is maintained, then it is expected that the wall temperatures would not reach the melting temperature of stainless steel and that the vessel integrity could be maintained. The thickness of any corium crust in contact with the vessel surface can be approximated as the solution of the general conduction equation with a volumetric heat source. This is given in Cartesian co-ordinates, consistent with the severe accident code MAAP, as the crust is assumed to be very thin and curvature effects are neglected:

$$\frac{d^2T_{crust}(x)}{dx^2} + \frac{q'''}{k_{crust}} = 0 \tag{8}$$

where

T_{crust} is the temperature of the crust

q''' is the volumetric heat generation rate in the crust

k_{crust} is the thermal conductivity of the crust

The vessel wall is however, solved separately utilizing radial co-ordinates consistent with the severe accident code MAAP. Assuming a local heat flux of q'' from the molten pool into the debris crust, the temperature at the inside surface of the vessel that is contacting the debris is given by:

$$T_i = T_o + \frac{q''r}{k_{ss}} \times ln\left(\frac{r+\tau}{r}\right) \tag{9}$$

where,

k_{ss} is the thermal conductivity of the vessel wall stainless steel

τ is the vessel thickness

r is the inner radius of the vessel

The temperature at the outer surface of the vessel T_o is estimated using the convective boundary condition, which involves the nucleate boiling heat transfer correlation (Roshenow type correlation for pool boiling) and the known heat flux at the outer boundary. The boundary conditions are: (a) at the interface of the molten corium pool and the crust, the temperature is equal to the corium liquidus temperature T_{liq}, and (b) at the interface of the crust and the inner wall of the vessel, the temperature is equal to the vessel wall temperature T_i. Assuming that the local heat flux from the molten pool is given by q'', the thickness of the crust must be such that at the interface between the molten pool and the crust, the following relation is satisfied

$$-k_{crust}\frac{dT_{crust}(x=0)}{dx} = q'' \tag{10}$$

The heat flux q'' is the downward heat flux from the molten pool, which varies azimuthally along the curved surface of the calandria vessel. The temperature distribution in the crust after incorporating the boundary conditions is given as:

$$T_{crust}(x) = -\frac{q'''x^2}{2k_{crust}} - \frac{q''x}{k_{crust}} + T_{liq} \quad (11)$$

The thickness of the crust is then given by:

$$\tau_{crust} = -\frac{q''}{q'''} + \sqrt{\left(\frac{q''}{q'''}\right)^2 + \frac{2k_{crust}}{q'''}\left(T_{liq} - T_i\right)} \quad (12)$$

For a crust thermal conductivity of 4.6 W/m-K (see Table 1) and liquidus temperature of 2673 K [19], the variation of the crust thickness along the vessel wall is shown in Figure 4 under ex-vessel cooling conditions (with nucleate boiling maintained)[3]. Due to the fact that CHF has not been exceeded, significant protective crusts are present. Even at the most limiting location, at the top of the debris bed where heat flux peaking occurs, sufficient crust thickness is present. Such crust evaluation methodology needs to be taken into account when assessing the vessel integrity as a whole. It is to be recognized that the presence of crust will change the heat transfer in the molten corium pool, in fact, the heat transfer rate goes down in the presence of a crust [3]. The heat transfer coefficient is about 10% lower for the situation when a solid crust surrounds the molten corium pool[4].

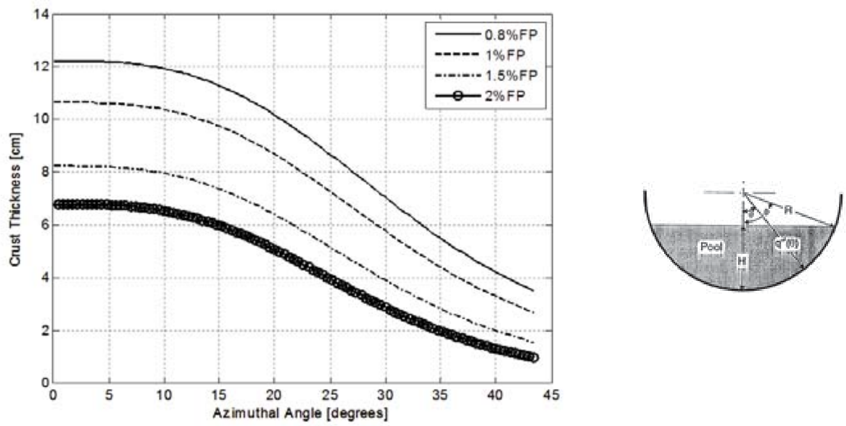

Figure 4: Variation of lower crust thickness with azimuthal angle

Narrow gaps can form between the crust and the inner vessel wall (due to the vessel wall being under tension and compressive loads), where water may be trapped to provide short to median term cooling effect[5] (observed from TMI-2 analysis), the superheat can basically vaporize the water in the narrow gap[6]. The CHF in the narrow gap is set by the CCFL and the heat flux limit corresponds to the latent heat of the water. Once the water is vaporized, steam trapped in the gap (if the steam does not egress out of the small area at the top edge of the debris bed) can provide cooling to the debris crust. Thus, under situations like this a temperature differential exists between the crust and the inner wall of the vessel, where the heat transfer initially is governed by boiling mode and later by a combination of conduction, radiation and local convection through the gas trapped in the gap. The gap effect is generally ignored when performing bounding analysis as the mechanistic understanding of the gap creation is not fully understood.

Given the above discussion on the characteristics of the corium pool, an assessment can now be performed to determine if IVR is achievable when ERVC is credited. The next section discusses the considerations that must be addressed in a Level 2 PSA to provide confidence in the likelihood of IVR.

---

[3] The legends in the figure represent fraction of full thermal power. The representative decay power is around 1% at the time of complete melt pool formation.

[4] The heat transfer coefficients depicted through Eq.s (4), (5), (6) and (7) are for when crusts are not present.

[5] This time frame is with regards to the late stages of the in-vessel accident progression.

[6] In severe accident analysis code like MAAP, the gap is usually taken in the order of microns.

4. DECOMPOSITION EVENT TREE

A Containment Event Tree (CET) provides a means of systematically tracking the progression of an accident after the onset of severe core damage and serves the primary goal in Level 2 PSA of assessing the ex-plant release of radioactive material. A CET is intended to model all physical or chemical processes that could challenge containment integrity and influence the release of radioactive material. The branch points[7] or nodal questions in a CET are organized in a chronological manner that allows for temporal sequencing of events in the progression of the accident. Discrete time frames in accident progression are introduced to capture distinct stages and significant changes in fission product release behavior. Treatment of specific phenomena via individual branch point questions can lead to very complex and large CETs. Where the branch point in the CET cannot be quantified directly, it may instead be decomposed into multiple factors that determine whether the event would occur to a level that can be quantified (e.g., phenomena, accident conditions, system or operator response). It is in this regard that a Decomposition Event Tree (DET) is developed that can be used to treat in more detail a simplified branch point. The factors are assembled in a chronological order in the form of a small "decomposition" event tree, or as a simple fault tree or logic diagram. The outcomes of a DET or the logic diagram are "rolled up" to define the split fraction for branches on the main event tree. A DET identifying the phenomenological challenges that could potentially lead to different RPV failure modes is shown in Figure 6. The DET is created based on the assumption that ex-vessel cooling is present and water contact with the outer wall of the vessel is always maintained, when the core debris accumulates at the bottom of the RPV. The end states as shown in the DET refer to the following:

- EARLY, FAIL: This represents the early failure mode of the RPV shell. This type of failure mode would occur if both internal and external cooling of the core debris is judged to not be effective.
- MEDIAN, FAIL: This represents a failure mode of the RPV shell during the re-melting process of the particulate core debris. This type of failure mode may occur given only external cooling is available to maintain the RPV integrity.
- LATE, FAIL: This represents a failure mode of the RPV shell when the coherent molten pool forms. This type of failure mode may occur given only external cooling is available to maintain the RPV integrity.

The aforementioned timings as associated with the RPV failure mode are relative to the overall progression during the in-vessel stage of the severe accident scenario. The DET assumes that if the challenges due to the in-vessel flooding do not pose a threat to the RPV integrity then IVR is assured. A brief description of the different challenges to the RPV in the absence of in-vessel flooding is provided below:

Does molten material attack fail the RPV shell?

The interaction of molten material with the RPV wall has the potential to lead to molten attack and erosion of the RPV steel. During the re-melt process of the particulate core debris, there is a potential of high temperature melt (molten Zr) relocating through the porous debris bed and contacting the stainless steel RPV wall and thus, consideration must be given to a potential chemical attack of the wall. The exothermic Fe-Zr chemical reactions can lead to eutectic dissolution of the RPV. Such chemical attack is strongly dependent upon the temperature of the melt and the wall and also upon the chemical composition of the melt[8]. Should the temperature of the interface[9] between these two materials (Fe-Zr) exceed the

---

[7] Branch points are questions to address whether specific phenomena occur in each time frame. The branch points are assigned probabilities based on analytical and qualitative assessments, the CETs are further integrated with Level 1 PSA fault trees to estimate the overall risk from a severe core damage accident scenario.

[8] It was observed in the TMI-2 post-accident analyses that essentially no attack of the vessel wall was observed as a result of the approximately 20 tons of oxidic material draining into the lower plenum.

[9] The contact of molten Zr with the vessel steel will lead to a variation of interface temperatures depending upon the

temperature required for eutectic formation, the exothermic nature of the eutectic reaction between Zr and Fe could lead to a runaway reaction that would cause a perforation of the RPV. Because this could happen in multiple locations as a result of the incoherent melting process of the core debris bed, this could effectively lead to multiple perforations of the RPV wall and a pathway for corium to relocate out of the vessel. However, the potential of such an attack can be considered to be minimal for the following reasons:

- Due to the presence of external cooling of the RPV, the outer regions of the RPV are maintained at a low temperature given nucleate boiling on the vessel outer surface. This stabilizes the system against chemical attack as the growth of a eutectic zone can be significantly limited should sufficient energy transport occur to limit the temperature excursion.
- It is not likely that in the highly heterogeneous mix of materials that pure Zr will be able to melt directly into interstitial regions of the porous bed. It tends to be confined or in direct contact with either UO_2 or ZrO_2 prior to undergoing melting. So it is more likely that any displaced melt will be a mixture of UO_2, Zr and ZrO_2. As well, this material will tend to be highly oxidic so there is limited potential for pure molten Zr impingement at this stage.

Due to the fact that transport of molten Zr through a porous debris bed will tend to promote freezing of the debris, it is not certain that molten Zr will contact the structural steel of the RPV. However, the complexity of material relocation through this porous debris bed makes it indeterminate as to whether molten Zr will contact the RPV structural steel. A subjective (conditional) probability should be assigned to represent the potential of molten Zr coming into contact with the RPV wall structural steel. Further, for a conservative analysis the initial interface temperature upon contact between molten Zr and vessel wall steel can be estimated by solving the transient conduction equation for a semi-infinite body with a constant surface temperature. The interface temperature[10] is given as:

$$T_s = \frac{\sqrt{k_c \rho_c c_c} T_{c,i} + \sqrt{k_{ss} \rho_{ss} c_{ss}} T_{ss,i}}{\sqrt{k_c \rho_c c_c} + \sqrt{k_{ss} \rho_{ss} c_{ss}}} \tag{13}$$

where
$T_{c(ss),i}$ is the initial temperature of the core debris (stainless steel) prior to contact,
$k_{c(ss)}$ is the thermal conductivity of the core debris (stainless steel) prior to contact,
$\rho_{c(ss)}$ is the density of the core debris (stainless steel) prior to contact, and
$c_{c(ss)}$ is the specific heat of the core debris (stainless steel) prior to contact

A conditional probability distribution should be assigned to represent the subjective range of Zr melt superheats. The onset of a rapid process of liquefaction is initiated at approximately 1523 K (Fe-Zr phase diagram suggests that eutectic points are 1201 K and 1601 K, with possibly lower eutectics as well) [22]. Due to the fact that energy transport occurs to the cooled side of the RPV wall, this will tend to prevent the significant liquefaction of the wall should the interface temperature remain below about 1523 K. Eventually the interface temperature between the debris and the RPV steel wall will decrease should sufficient cooling be present. Since the initial contact between the molten material and RPV steel wall will occur with ex-vessel cooling present, sufficient heat removal will be available. An overall judgment should be made and appropriate probability should be assigned to this branch point.

Does molten material attack fail the RPV penetrations?

This nodal question addresses the issue of molten corium relocating into penetrations such as In-core Instruments (ICI), control rods etc. This is treated in the DET as a threat to the RPV integrity during the

---

degree of Zr superheat.

[10] Over the initial transient, this interface temperature can be expected to be constant since cooling of the molten material will occur via heat conduction into the cooler RPV steel.

re-melting of the core debris. The physical issue relates to the fact that the melt entering the penetrations can cause the penetrations to heatup and lead to weld failure and/or melt ejection through the penetration thereby, compromising the penetration integrity and leading to RPV failure. The penetrations are usually concentric tubes with water filled annulus and welded to the RPV lower head. A conservative assessment should be performed to ascertain the wall temperatures of the tubes by solving for the steady state conduction heat transfer in a concentric cylinder with volumetric heat source generation, with appropriate outer boundary conditions. The volumetric heat source should conservatively correspond to decay heat levels at the melt relocation times into the lower plenum of the RPV[11]. For example, in a severe accident scenario (with ex-vessel cooling) the temperature profile across the penetration surface is shown in Figure 5 with nucleate boiling maintained on the outside of the penetration. The penetration in this case will not undergo melt-through. Additionally, a weld integrity assessment needs to be performed under such a relocation scenario, keeping in perspective the conclusions drawn from observations made from the TMI-2 accident and KAERI experiments [3] on penetration tube failure mechanism. An overall judgment should be made and appropriate probability should be assigned to this branch point.

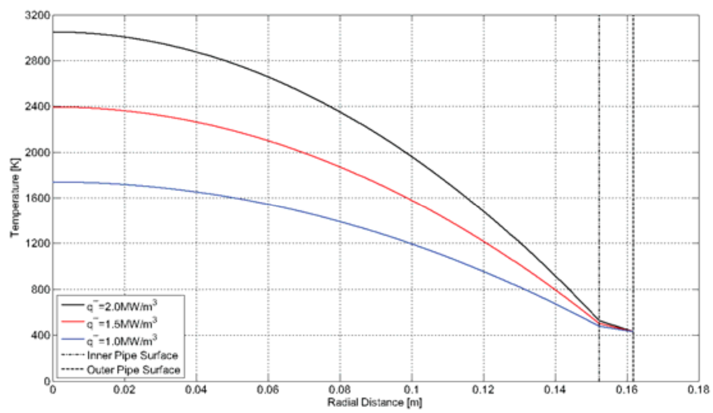

Figure 5: Temperature profile of corium in a penetration

The following three nodal questions deal with the late failure modes of the RPV:

Does molten material stratify into oxidic and metallic debris leading to failure of the RPV?
This nodal question pertains to the metal layer stratification issue and the threat it poses to the RPV structural integrity. This issue was discussed in detail in Section 3.1, an assessment needs to be performed accordingly with uncertainty bands taken into consideration.

Is the heat flux through the RPV shell above the CHF?
This nodal question pertains to the issue of through wall heat flux for the RPV shell from the molten corium debris pool. This heat flux should be lower than the CHF so as to avert the boiling crisis and prevent the RPV failure. Further, the wall thickness of the RPV shell should be such that it is greater than the CHF limited wall thickness so as to successfully carry the load imposed by the dead weight of the core debris. This issue, of the through wall heat flux, is based on the specifics of the molten corium pool that is described in detail in Sections 0 and 3.3 respectively.

---

[11] The DET assumes this challenge during the re-melting of the core debris in the lower plenum, the decay heat levels will be minimally different.

Does long term corrosion lead to vessel failure?

Another physico-chemical issue that can potentially challenge the vessel integrity in the long term is the corrosion of the vessel wall due to the corium debris/crust attack. The issue related to corrosive attack of corium on the pressure vessel has been considered in detail in References [23] and [24] under the METCOR project for the VVER reactors. The study was performed for the IVR scenario under ex-vessel cooling. The essential drivers for steel corrosion are: a) surface heat flux, b) steel-corium interface temperature, c) corium melt composition (oxidized corium/sub-oxidized corium) and, d) oxidizing/inert atmospheric conditions. The main conclusion from this study was that under ex-vessel cooling where there is a large temperature gradient, surface corrosion will not jeopardize the mechanical integrity of the vessel wall because the load carrying capacity will be borne by the outer surface of the vessel, which is cooled. However, for each reactor type the situation may be unique and therefore, a holistic view of the problem needs to be considered before making a judgement. Further, it is necessary to investigate those scenarios where external cooling to the RPV vessel wall is eventually depleted.

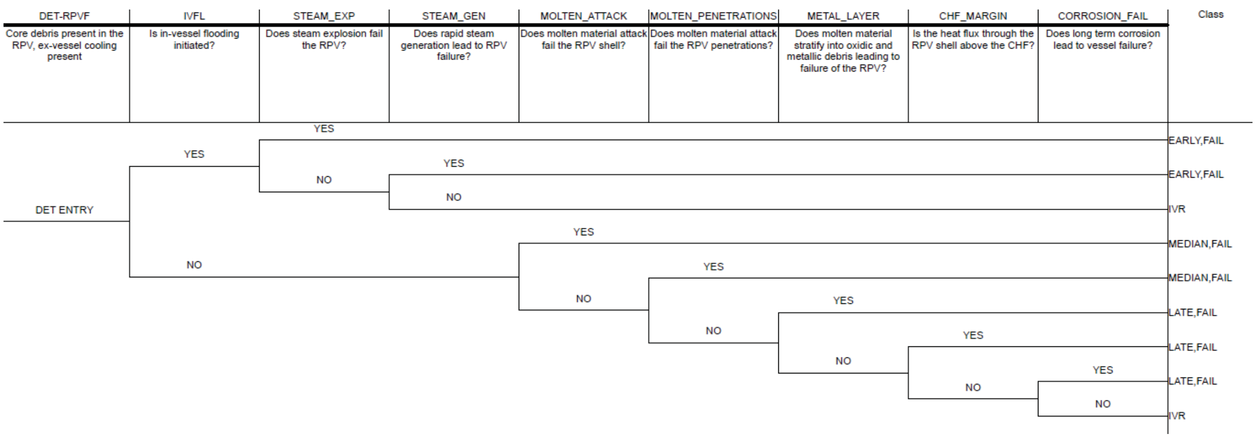

Figure 6: Decomposition Event Tree for RPV failure modes in the presence of ERVC

5. CONCLUSIONS

This paper identified the physical behavior associated with the molten corium pool. Furthermore, it discussed the ensuing thermo-chemical-mechanical challenges that the molten corium pool poses to the integrity of the Reactor Pressure Vessel (RPV) lower head under External Reactor Vessel Cooling (ERVC) and how to address these challenges from a Level 2 PSA perspective. A DET was presented in this study for use in Level 2 PSA analysis. Appropriate bounding analysis needs to be performed for the scenarios identified in the DET and branch probabilities should be assigned based on subjective engineering judgment.

REFERENCES

1. T.G. Theofanus, C. Liu, S. Addition, S. Angelini, O. Kymäläinen and T. Salimassi, "In-Vessel Coolability and Retention of a Core Melt," *DOE/ID-10460*, **Vol. 1&2** (1996).
2. S.I. Pantyushin, E.A. Frizen, V.P. Semishkin, N.V. Bukin, M.A. Bykov and V.A. Mokhov, "Consideration of a Possibility for Corium Retention (Reactor Internals and Core Melt) in the Vessel of WWER Reactor from 600 to 1300 MW," *5th European Review Meeting on Severe Accident Research (ERMSAR-2012)*, Cologne, Germany, March 21-23, 2012.
3. R.J. Park, K.H. Kang, S.W. Hong, S.B. Kim and J-Ho. Song, "Corium Behavior in the Lower Plenum of the Reactor Vessel under IVR-ERVC Condition: Technical Issues," *Nucl. Eng. Tech.* **44**(3), pp. 237-248 (2012).

4. J.L. Rempe, K.Y. Suh, F.B. Cheung and S.B. Kim, “In-Vessel Retention – Recent Efforts and Future Needs,” *The 6th International Conference on Nuclear Thermal Hydraulics, Operations and Safety (NUTHOS-6)*, Nara, Japan, October 4-8, 2004.
5. R.J. Park, K.H. Kang, K.I. Ahn and S.W. Hong, “Analysis of Corium Behavior in the Lower Plenum of the Reactor Vessel during a Severe Accident,” *5th European Review Meeting on Severe Accident Research (ERMSAR-2012)*, Cologne, Germany, March 21-23, 2012.
6. T.N. Dinh, J.P. Tu and T.G. Theofanus, “Two-Phase Natural Circulation Flow in AP-1000 In-Vessel Retention Related ULPU-V Facility Experiments,” *ICAPP’04*, Pittsburgh, PA, USA , 2004.
7. S. Rouge, I. Dor and G. Geffraye, “Reactor Vessel External Cooling for Corium Retention SULTAN Experimental Program and Modelling with CATHARE Code,” *Workshop Proceedings on In-Vessel Core Debris Retention and Coolability*, NEA/CSNI/R(98), 1998.
8. B.R. Sehgal et al., “Assessment of reactor vessel integrity (ARVI),” *Nucl. Eng. Des.* **221**, pp. 23-53 (2003).
9. G. Repetto, T. Garcin, M. Rashid, R. Kulenovic, W. Ma and L. Li, “Investigation of Multidimensional Effects during Debris Cooling,” *5th European Review Meeting on Severe Accident Research (ERMSAR-2012)*, Cologne, Germany, March 21-23, 2012.
10. S. Leininger, R. Kulenovic, S. Rahman, G. Repetto and E. Laurien, “Experimental investigation on reflooding of debris beds,” *Annals of Nuclear Energy* **74**, pp. 42-49 (2014).
11. W. Klein-Heβling et al., “Conclusions on severe accident research priorities,” *Annals of Nuclear Energy* **74**, pp. 4-11 (2014).
12. Executive Summary. CSNI/NEA RASPLAV Seminar 2000, Munich, Germany, November 14-15, 2000.
13. M. Fukasawa and S. Tamura, “Thermodynamic Analysis for Molten Corium Stratification Test MASCA with Ionic Liquid U-Zr-Fe-O-B-C-FPs Database,” *Nucl. Sc. Tech.* **44**(9), pp. 1210-1219 (2007).
14. J.M. Bonnet and J.M. Seiler, “Thermal Hydraulic Phenomena in Corium Pools: The BALI Experiment,” *7th International Conference on Nuclear Engineering (ICONE-7057)*, Tokyo, Japan, 1999.
15. A. Palagin et al., “Analysis and interpretation of the LIVE-L6 Experiment,” *5th European Review Meeting on Severe Accident Research (ERMSAR-2012)*, Cologne, Germany, March 21-23, 2012.
16. G. Balog, “Research and Development in Support of Beyond Design Basis Accidents (BDBA) Response in CANDU Nuclear Power Plants,” *IAEA International Experts Meeting 8*, Vienna, Austria, February 2015.
17. J.M. Seiler, B. Tourniaire, F. Defoort and K. Froment, “Consequences of material effects on in-vessel retention,” *Nucl. Eng. Des.* **237**, pp. 1752-1758 (2007).
18. J.M. Seiler and K. Froment, “Material Effects on Multiphase Phenomena in Late Phases of Severe Accidents of Nuclear Reactors,” *Multiphase Science and Technology.* **12**, pp. 117-257 (2000).
19. A.P. Muzumdar, P.M. Mathew, J.T. Rogers and M.L. Lamari, “Core Melt Retention Capability of CANDU Reactors,” *Canadian Nuclear Society Conference*, Banff, AB 1998.
20. J.L. Rempe, D.L. Knudson, C.M. Allison, G.L. Thinnes, C.L. Atwood and M.J. Cebull, “Potential for AP600 In-Vessel Retention through Ex-Vessel Flooding,” *Technical Evaluation Report INEEL/EXT-97-0079*, 1997.
21. X. Gaus-Liu, A. Miassoedov, T. Cron, S. Schmidt-Stiefel and T. Wenz, “The Experimental Results of LIVE-L8B – Debris Melting Process in a Simulated PWR Lower Head,” *5th European Review Meeting on Severe Accident Research (ERMSAR-2012)*, Cologne, Germany, March 21-23, 2012.
22. P. Hoffman, S.J.L. Hagen, G. Schanz and A. Skokan, “Reactor Core Materials Interactions at Very High Temperatures,” *Nucl. Tech.* **87**, pp. 146-187 (1989).
23. S.V. Bechta et al., “Corrosion of vessel steel during its interaction with molten corium Part 1: Experimental,” *Nucl. Eng. Des.* **236**, pp. 1810-1829 (2006).
24. S.V. Bechta et al., “Corrosion of vessel steel during its interaction with molten corium Part 2: Model Development,” *Nucl. Eng. Des.* **236**, pp. 1362-1370 (2006).