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REVIEW OF

THE AGEING PROCESSES AND THE INFLUENCE ON SAFETY AND PERFORMANCE OF

WYLFA NUCLEAR POWER STATION

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THE AGEING PROCESSES AND THE INFLUENCE ON SAFETY AND PERFORMANCE AT WYLFA

SUMMARY

This Review considers how ageing of the Magnox nuclear power station at Wylfa could be expected to influence performance and safety.

For the Wylfa reactors, the ageing processes apply to a diverse range of different materials and components. Some of these ageing processes are relatively straightforward and well understood; others are complex and have yet to be fully understood. As time passes, it becomes increasingly more difficult, if not more unreliable, to predict the types of age-related problems that are likely to be encountered now and in future years. In fact, as the reactors move well beyond the 20 to 25 year design life originally specified, a greater reliance has to be placed on inspection of in-reactor materials and components and, from this, the ageing effects deduced. The problem here is that the Wylfa reactors do not include features that enable ready access to all of the components susceptible to ageing.

Even once identified, it may be difficult to establish how the age-related degradations might apply to the plant overall during normal operation and, particularly, when the plant is under fault conditions. Indeed, ageing may introduce aspects of plant performance and response that were unforeseen by the plant's original designers and for which they provided no contingency.

Three specific ageing effects are examined. These are the cracking of the reactor pressure vessel steel liner closure plates in the vicinity of the vessel wall penetrations carrying the superheated steam tailpipes from each boiler; the corrosion of the internal steelwork of the reactors, particularly the core restraint garter; and the radiolytic oxidation (corrosion) or loss of volume of the graphite core. The Review examines how each of these ageing effects might contribute to reactor fault conditions, particularly where the simultaneous failure of a group of superheater tailpipes results in high pressure differentials within the reactor and which subjects the graphite core and its restraint system to excessive loading.

Excessive loading of the core structures could result in core misalignment. Once the core has been misaligned or damaged, the circumstances that could lead to localised overheating of fuel channels are examined in terms of the effectiveness of the primary circuit cooling plant to extract both the post trip decay heat and the release of heat from stored (Wigner) energy in the graphite core. For this case the detrimental influence of the steady build-up of carbonaceous dust over past years of operation, associated with graphite radiolysis, accumulating in and partially blocking the secondary and cross flow passages of the core is considered to contribute to localised overheating of the core. For the case where the pressure vessel containment has failed, for which the decay heat extraction must be completed with an open primary circuit with the core immersed in air, the additional contribution of Wigner energy, the increased chemical reactivity (burning) of the graphite and carbon dust are all considered to contribute to a deteriorating thermal situation, resulting in fuel temperatures sufficiently high to prompt magnesium clad and fuel ignition.

Importantly, the ageing of critical and essentially non-serviceable components within the reactors at Wylfa determines how these components perform under fault conditions. The original designers of Wylfa did not foresee and account for this ageing so the outcome of the

so called *Design Basis Accident* was based on certain components surviving unscathed during the then nominated worst case fault conditions. There is now considerable doubt that the graphite core could survive both rapid reactor depressurisation and steam intrusion fault conditions so the *Design Basis Accident* and its limited consequences, both of which continue to be adopted by the present operator and the regulator the Nuclear Installations Inspectorate (NII), are no longer valid.

Both reactors at Wylfa have been shut down since the discovery of the closure weld cracking in April 2000. Because the consequences of a single closure weld failure could present a *beyond design basis event* and trigger failure of the weakened core restraint system and distortion of the graphite core, BNFL Magnox's strategy of returning the reactors to power with an interim fix (see footnote 28) whilst the closure weld studies are underway, should be considered unacceptable because it continues to rely upon the integrity of the core restraint and core assembly system which, for the aged reactors at Wylfa cannot be stated with certainty.

The fact that there has been little modification to the fault conditions that make up the *Design Basis Accident* is particularly surprising since the NII has known of the deteriorating ageing conditions within the reactors since a year before Wylfa was first commissioned.

This is because two years before Wylfa was scheduled to start its nuclear reactors it was discovered that the steelwork internals of the other Magnox reactors, particularly at Dungeness and Bradwell, were corroding at an unacceptably high rate. In late 1970, less than year before criticality of the first of Wylfa's reactors, it was decided, at Cabinet Office level, that it would not be economic to rip out and replace the internal steelwork to inhibit corrosion but, instead and to slow the corrosion rate, the reactors would be temperature derated and the quality of the coolant gas modified. The effect of this latter modification was not fully understood at the time, particularly how the rate of graphite radiolysis increased markedly at the higher gas pressure of the Wylfa reactor circuits. The outcome was that, at the cost of slowing the steel corrosion rate, there resulted an increased rate of radiolytic loss (oxidation) of the graphite and a structural weakening of the graphite core assembly, both of which have significant safety implications.

The role of the NII is of interest in that it was criticised at the time of the discovery for its relationship with the then operator the CEGB (Central Electricity Generating Board) and that the problem had not been recognised as soon as it might have been. The NII has never acknowledged that the steel-graphite corrosion trade-off at Wylfa arose from its own recommendations of 1970 nor, in its reporting of subsequent years, has it indicated that the two processes are linked. Moreover, the NII has been slow to acknowledge the importance of loss of strength of the reactor cores due to graphite radiolysis linked to the deteriorating strength of the restraint garter. It was not until 1995 that it required the introduction of greater diversity in the reactor shut-down systems to cater for the greater potential of core distortion under its weakened condition, and as late as 1998 it noted that changes to material properties of the core at Oldbury power station were "subject to uncertainty", from which it might be assumed that the safety case for the core could never have been rigorously examined. The now abandoned proposal to deploy the enriched MagRox fuel at Wylfa, which was intended to compensate for the reduction in thermal moderation linked to graphite loss, also suggests that the NII had failed to grasp the extent by which this ageing process had depleted the moderating, and hence the strength of the reactor core.

In effect, when in 1971/2 the reactors at Wylfa were first brought into operation, there was considerable doubt and uncertainty about the future performance of crucial, in-reactor components as these aged. Put another way, if the basis of the design was that the reactor would survive the *Design Basis Accident*, then departure from the original design by unforeseen ageing processes would invalidate the *Design Basis Accident.*

Uniquely, Wylfa was a nuclear plant that was to be licensed in the knowledge that its safety margins would deteriorate over time in a manner and to an extent not foreseen by its designers. Thirty years after the commissioning of these reactors, the NII continues to express doubts as to the actual condition of the graphite core and its restraint garter and, in the view of the very limited inspection access, it continues to rely upon the operator to substantiate the safety case with further information drawn from ongoing studies. In this important respect the regulatory regime at Wylfa seems to be reactive rather than prescriptive.

Why the NII has chosen never to declare that it knew, from the onset, that there were serious ageing problems underway at Wylfa is baffling. The NII's Long Term Safety Review for Wylfa (1995) reports both steel and graphite corrosion in a matter of fact way, implying that the steel oxidation is 'well understood and managed". This is entirely in contrast with its startling discovery in 1969-70, which called for a decision on whether to strip out the incorrectly specified steels from both Wylfa reactors before their respective start ups which would render the reactors radioactive thus precluding any major modifications in future years. We now know that the reactors were started without modification and that this decision, taken at the highest political level, was in line with the NII's recommendation of that time.

Finally, the fact that the NII not only knew but, indeed, was instrumental in the 1970 decision to put Wylfa into service without modification puts a whole new light on the nuclear safety regulatory regime as then practised in the UK.

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PART I – DESIGN AND OPERATION OF MAGNOX NUCLEAR POWER STATIONS

1) WYLFA MAGNOX NUCLEAR POWER STATION

There are three distinctive design types of Magnox nuclear power station.

The steel pressure vessels in the earliest power stations at Calder Hall and Berkeley are cylindrical, in the later stations built during the so-called reactor *'baby boom'* of the 1960s the steel pressure vessels are spherical, and for the last two stations at Oldbury and Wylfa the pressure vessels are of pre-stressed, reinforced concrete construction:

TABLE 1 MAGNOX POWER STATIONS

STATION	YEAR OF COMMISSION	CAPACITY DESIGN MW_e [2]	PRESENT DERATED MW_e	TYPE	COMMENTS	LTSR DATE
Calder Hall	1956 - 59	4 x 60	4 x 50	Cylinder steel	Operational	1990 – 34 yrs
Berkeley	1961 - 62	2 x 167	-	Cylinder steel	Defuelled – part dismantled	1988 – 27 yrs
Chapelcross	1958 - 60	4 x 60	4 x 50	spherical steel	Operational	1990 – 32 yrs
Bradwell[1]	1962	2 x 150	2 x 121	spherical steel	Operational	1987 – 25 yrs
Hinkley Point A	1965	2 x 265	(2 x 235)	spherical steel	Closed - May 2000	1991 – 26 yrs
Trawsfynydd	1965	2 x 236	-	spherical steel	Defuelled – part dismantled	1993 - 28 yrs
Dungeness A	1966	2 x 228	2 x 219	spherical steel	Operational	1994 - 28 yrs
Hunterston A	1964	2 x 169	-	spherical steel	Defuelled – part dismantled	1989 – 25 yrs
Sizewell A	1966	2 x 250	2 x 210	spherical steel	Operational	1995 – 29 yrs
Oldbury	1968	2 x 225	2 x 217	RC Tendon	Operational	1995 – 27 yrs
Wylfa	1971	2 x 570	2 x 490	RC Tendon	Operational	1995 – 24 yrs

NOTES: 1 There are identical single reactor stations at Tokia Muria Japan and at Latina in Italy, both are closed down and have been defuelled.
2 Present power ratings are given although all of the steel pressure vessel reactors have been derated over the years.
3 Magnox Generation (British Nuclear Fuels) has recently announced closure dates for all of the steel RPV Magnox power stations, with all of these stations shutting down within the present decade. In fact a number of these power stations have not operated for some time, being held on extended outages and it is believed to be unlikely that the Magnox power stations at Hinkley Point and Bradwell will recommence commercial generation again.
4 Trawsfynydd was closed down before its LTSR was completed, although a summary of the LTSR was published in 1993

The Magnox nuclear power stations at Oldbury and Wylfa include a number of significant design departures from the earlier Magnox reactor designs. Essentially, the Oldbury/Wylfa pressure vessels are of massive, reinforced concrete, compared to the steel shells of all other units; the steam raising boilers are located within the main pressure vessel; and the coolant gas operating pressure is higher. The last of these two power stations to be constructed, Wylfa, is much larger than any of the previous Magnox units with each reactor of a design rating of $570MW_e$.

In summary: The last two Magnox power stations designs are considered to have provided the stepping stone between the earlier Magnox in-steel reactors to the next generation of nuclear power stations, the advanced gas-cooled reactors (AGR), which adopted similar reinforced concrete pressures vessel technology with the boilers located within the pressure vessel containment. The reactor power rating of the Wylfa reactors match the size of the AGR reactors with this, and other features of the nuclear and steam raising plant, are reckoned to have played a significant role in proving the AGR design.

2) THE SAFETY REVIEWS

Before a nuclear power station is permitted to commence operation it must be licensed by the Nuclear Installations Inspectorate (NII).[1] The licensing requirements relates to the health and safety of persons and property, primarily centred around the performance of the reactor safety and containment systems during and following adverse (accidents) events.[2]

The approach to assessing nuclear safety and the criteria defining an acceptable safety regime have developed considerably since the design of Wylfa in the 1960s. The NII endeavours to take account of differences between the original and the modern safety standards by its *Safety Assessment Principles* (SAPs)[3] which includes the overriding principle that all risks must be *as low as reasonably practicable* (ALARP), which is applied via a *probabilistic safety analysis* (PSA or PRA).

Difficulties can arise in the application of the SAPs and PSA to earlier nuclear plants such as Wylfa. First, the original design may not amenable to this approach and, second, ageing processes may have introduced other mechanisms for failure and abnormal fault conditions.

At Wylfa, for example, the resistance to seismically induced loading of the original reactor core design and installation does not meet modern requirements and there is a possibility that during an earthquake the core could distort and lose alignment with the control rods that are required to close down the reactor. However, since upgrading the core structure is not practicably possible, the c1995 compromise of installing a number of articulating control rods that can snake through a seismically distorted core is accepted as being an ALARP solution, although this approach would not be accepted in a modern nuclear plant to provide a truly 'diverse' close down system.

1 The NII is a division of the Health and Safety Executive that regulates and licenses nuclear installations under the Nuclear Installation Act 1965.

2 The Act stipulates that no injury to persons of damage to property is caused by radiation arising from the power station site and an absolute liability is placed upon the licensee to secure this. There is a similar authorisation and certification process under the radioactive Substances Act 1960 overseen by the Environment Agency that relates to discharges of radioactive substances.

3 Safety Assessment Principles for Nuclear Plants, NII, HSE, 1992

Taking this example further, in the original design of Wylfa a seismically induced core distortion was considered to be a *beyond-design-basis* event (ie it would not happen) whereas today, the more demanding seismic specification, core distortion has to be accepted to be a within *design-basis* event (ie foreseeable and could happen). Again today, it would be very unlikely that a new-build nuclear plant, equivalent to a Magnox, would be licensed for operation if the core was at risk of distortion and, particularly, if there was no diverse means of shut-down (ie the articulated control rods are not a sufficiently diverse because for operation these are also at risk to core distortion).

The ability of the power station to meet the safety criteria stipulated in the licence was periodically reviewed but in the late 1980s, by which time some of the earlier steel pressure vessel types were approaching or exceeding the useful, safe life of 20 years[4] of being in service, the NII required the operator to assess the *Generic Safety Issues*, particularly those developing with the ageing of the reactors.[5] The NII also required the operator to prepare a safety review for each specific nuclear power station, which became known as the *Long Term Safety Review* (*LTSR*), before that power station could continue operating past its 20 year life.

In or about 1995 the NII added a further constraint to the Magnox power station licenses by requiring those power stations approaching 30 years of operation[6] to undergo a *Periodic Safety Review (PSR)* before that station would be permitted to continue in operation beyond the 'milestone' of 30 years, although as can be seen from TABLE 1 a number of plants have overshot this target.

It follows that the LTSR, prepared by the operator, is a key staging point for the continued safe operation of each individual nuclear power station. Surprisingly, no part of this LTSR is publicly available,[7] instead the NII publishes its assessment of the operator's LTSR in summarised form that are of limited value since little data and 'hard' information is presented.

For example, in the three areas of ageing reviewed here, in 1995 the NII LTSR reported for Wylfa:-

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4 There is no doubt that the station amortisation life was 20 years and this would have been the overall time period specified to the engineering teams undertaking the design work.

5 Magnox Nuclear Power Programme, NII's Report on the Outcome of the Programme of Work on Generic Safety Issues, NII, HSE, 1994

6 For example, NII Press Release, 25 October 2000 – the 1995 LSTR states that the NII has concluded that the reinforced concrete pressure vessels at Wylfa were then safe to continue in operation to an age of 33 years (2004), although there is no explanation as to why the life expectancy is such an odd number (ie 33 years and not 30 years).

7 The NII state this is because the information contained in the LTSR is owned by the operator and for reasons of *commercial confidentiality* it cannot be released by the NII.

Steel Liner to Pressure Vessel

“. . . A structural integrity case has been presented for all penetrations whose failure could lead to a beyond design basis event. This was based on good quality design and installation, comparison with modern standards, their limited exposure to degradation mechanisms, proof pressure testing and inspection programmes which are also monitored by the NII. Although we judge that the penetrations are adequately safe for continued operation we have asked NE (Nuclear Electric) to undertake a programme of follow up work to provide additional justification of their claims. . . . We are therefore satisfied that NE have demonstrated that the PCPV liner insulation is fit for continued service up to at least 33 years. . . .”

Graphite Core

“. . . We have reviewed NE’s ongoing surveillance programme of graphite monitoring and we accept that it addresses the relevant chemical and physical properties. NE have used the predicted graphite properties in presenting their structural integrity case and depressurisation fault studies for continued safe operation of the reactors up to at least 33 years . . .”

Steel Core Restraint

“. . . The core restraint is a complex system which is essential to maintain the overall stability of the reactor core. No facility for in-service inspection was provided in the original design. Consequently NE’s review has concentrated on the original design safety case and considered predicted effects associated with oxidation, creep, fatigue, irradiation, common mode failure and fault conditions. Arising from this NE have in hand a programme of work to provide even further assurance that the core restraint should remain safe during operation up to at least 33 years. . . .”

These three examples illustrate the just how little hard information is presented by the NII in accepting the operator’s LTSR. It is not possible for third parties to form any meaningful opinion as to how the operator demonstrated to the NII that these reactor components were fit for a further (then) ten years of operation (at which a total life of 33 years would have been reached). Indeed, the NII summaries include much ambiguity - what is meant *‘adequately safe’* - generalities - what are the relevant *‘chemical and physical properties’* - and, more often than not, these are open-ended - what is the *‘programme of work’* which is to provide *‘even further assurances’* and when will this programme be completed ?

In Summary: Considerable delays occurred in the Magnox stations completing the LTSRs to the 20 year design life threshold, with most LTSRs failing to be published before the relaxed target date of 25 years (see TABLE 1).

For the Oldbury and Wylfa power stations the original working life is believed to have been specified at 20 years,[8] which seems to have been the target period adopted for their LSTRs. The 30 year milestone PSR for Oldbury was published on the date originally expected in 1998, but the PSR for Wylfa is not planned to be available until 2004.[9]

The NII's assessment of the safety case for nuclear plants such as Wylfa, via its SAPs, includes opportunity for the safety case to be approved when, in fact, it falls far short of modern expectations. The NII claims that it is not necessary for older plants to be capable of full compliance with current safety standards and that such requirements are not absolutely necessary from a regulatory standpoint providing, that is, an acceptable case exists on the basis of *"engineering judgement"*.[10]

The LTSR reviews published by the NII are not informative in a technical sense. Although the LTSR might inform the public of the decisions reached by the NII, it does not reveal the detail of how it reached these decisions. This is important because there is no sense of the detail of the NII's checking of the safety case, how rigorous its assessment was and, importantly, the emphasis that it, the NII, placed on certain aspects and components of the reactor. We now know, for example, that the reactor pressure vessel liner cracking at the superheater closure penetrations was not detected in the 1995 LTSR, although it has now emerged as a major fault.

Also, as this Review will consider later, neither the generic safety issues studies or the LTSR refer to the discovery of accelerated corrosion of the reactor steelwork in 1969 and that, particularly for Wylfa, this was to have very significant performance and safety implications for its entire working life.

8 The NII give the 'minimal' working life to be "of about 20 to 25 years" – see Wylfa Nuclear Power Station, The Findings of NII's Assessment of Nuclear Electric's Long Term Safety Review, NII, HSE, 1995 - although such ambiguity is very much against the deterministic design approach that was adopted for high technology applications during the 1960s when Wylfa was under design. In 1979 the then-operator, the CEGB, stated in a 1979 internal document that *"Operations Department advise that it is prudent to assume that all Magnox plant will have a 25-year life except for Wylfa where a 20-year life should be assumed because of higher gas coolant pressure causing increased steel and graphite damage."* 1979/80 Development Review, CEGB Planning Department

9 NII Press Release, 25 October 2000 – the 1995 LSTR states that the NII has concluded that the reinforced concrete pressure vessels at Wylfa were then safe to continue in operation to an age of 33 years (2004), although there is no explanation as to why the life expectancy is such an odd number (ie 33 years and not 30 years). Apparently, according to the NII, this arises because Wylfa's LTSR uniquely set a 35 year life when it was in its 23rd year, but the NII deemed the LTSR good for a further 10 years, hence the rather odd 33 year milestone life.

10 See Reference 5 – others might argue that this might be acceptable if the full details of the safety case were to be made available to the public so that others might share in the *'engineering judgement'* relied upon by the NII. In terms of the uncertainty of the ageing of certain in-reactor components, namely the graphite core and its restraint system, arriving at a *deterministic* engineering judgement cannot be justified.

3) OPERATION

Wylfa nuclear power station consists of two gas-cooled, natural uranium fuelled, graphite moderated reactors. Each reactor contains approximately 49,000 fuel elements, totalling about 600 tonnes of uranium, arranged in stacks of 8 inside individual fuel channels with the graphite moderator core which comprises about 3,800 tonnes of graphite. The gas containment is within a reinforced concrete pressure vessel with a spherical internal void, of approximately 30 meters diameter, housing the reactor.

Carbon dioxide gas is circulated up through the fuel channels, heated by the nuclear fission process underway in the fuel and the channelled into boilers where steam is raised in a separate circuit that drives the turbo-alternators. The carbon dioxide cools, passes through gas circulators and delivered to the underside of the reactor core for reheating.

APPENDIX I provides a fuller description of the Wylfa reactors.

4) THE INFLUENCE OF AGEING FACTORS ON SAFETY

In the event of abnormal operations or fault conditions occurring there are a number of essential actions required from the plant operator. First, the nuclear reaction must be terminated; second the residual and decay heat of the reactor core and nuclear fuel must be dissipated; and, third, throughout these two processes the containments, both fuel cladding and reactor primary pressure circuit, must be maintained.

Reactor Shut-down Systems

As previously discussed, the reactor shut-down system must be reliable under all foreseeable fault conditions. For this, the means of shut-down must have considerable redundancy and comprise sufficiently diverse means.

At Wylfa there are two shut-down systems for each reactor.

The operational shut-down system is insertion of the control rods into the core channels, with this system being augmented by a number of articulated control rods that can descend into channels that may have misaligned due to core distortion. Distortion of the core may be a result of the fault condition underway at the time that emergency shut-down is required. Although the articulated control rods provide a degree of diversity, other than being able to cope with a limited degree of core distortion these rods share that same features and prerequisites required for operation for the other control rods (clutch release and standpipe route integrity).

The second shut-down system is whereby boron dust is injected into the core channels to suppress neutron activity and hence quench the chain reaction. This terminal system is also not fully diverse because it also relies upon the channels in reactor core remaining accessible and on continuing coolant gas flow to fully disperse the neutron absorbing dust.

Residual and Post Trip Decay Heat Removal

In the period immediately following a reactor shut-down, since the reactor core and nuclear fuel hold a very significant quantity of heat (by virtue of the large thermal mass) it is vital to maintain boiler water supplies to remove this heat. In addition to this 'stored' heat, the nuclear fuel continues to undergo the radioactive decay process which, alone and in the absence of continuing criticality, generates additional heat at about 10% of full reactor power for the first 30 or so minutes, thereafter decaying over the next few hours and days as the short-lived radioisotopes naturally decay.

At Wylfa there are secondary and tertiary feedwater supplies to the boilers that provide diversity and standby gas turbine generators are available should electrical supplies to the feedwater pumps be lost. If the reactor circuit remains pressurised with carbon dioxide then there is adequate heat dissipation capacity in the system for the reactor to post-trip cool on natural circulation once that the gas circulators have spun down. If the circuit is depressurised but contained, then the gas circulators have to be powered throughout the post trip period.

If, however, the reactor core has distorted, fuel elements within the fuel channels could be denied cooling gas and localised overheating could lead to fuel clad ignition. If the reactor circuit is breached and if air is present, this overheating could result in a uranium fuel fire.

Pressure Vessel Containment Integrity

Generally, it is acknowledged that catastrophic failure of the reinforced concrete pressure vessel is unlikely. However, the whole containment boundary includes potential failure sites such as the individual fuel and control standpipes that lead from the reactor pile cap floor into the reactor; the ducting that conveys coolant to the automatic pressure relief valves; the automatic pressure relief valves themselves; the gas circulators; and each of the numerous services penetrations that pass through the walls of the reinforced concrete pressure vessel, including the boiler feedwater supplies and steam superheater outlets.

The reactor component that must remain reliable for both shut-down and post trip fuel cooling is the graphite moderator core. At Wylfa the original design specification for the graphite core, comprising a loose keyed assembly of graphite blocks and the peripheral steel restraint garter was considered sufficiently robust to withstand all of the credible fault scenarios and, deriving from this, no facility was included for servicing and replacement of the core and restraint garter components.

Yet it is these critical components have been subject to ageing degradation.

At Wylfa (and Oldbury) the radiolytic oxidation is expected to exceed all other Magnox reactor graphite losses by the time that these reactors reach the end of their service lives – at Oldbury the recorded weight loss (via radiolytic oxidation) was measured at 12% after 18 years of operation and this station and Wylfa should have now lost about 20% if not more graphite in the most effected bricks (mid-core/mid-height).[11]

Weight loss of the graphite has two important outcomes: First, it reduces the amount of moderation available so it may be necessary to offset this by slightly enriching the fuel (ie increase the number of fast neutrons available) and, second, with weight loss there is a corresponding reduction in strength that may be significant in certain fault conditions where the inherent strength of the graphite assists the core restraint garter resistance to core misalignment and movement.

At Wylfa the original design considered failure of the core restraint garter to be a *beyond design basis event*, that is it was considered to be such a remote chance that it could be discounted (ie incredible). However, the practice of methane injection to inhibit graphite radiolysis is acknowledged to have induced corrosion of the core restraint components to the extent that the core restraint garter may no longer be relied upon as a failsafe assembly in the *design basis event*.[12]

Age-related factors relating to the garter include corrosion of the components and hence loss of strength; seizure of moving parts by the growth of oxide films (rust) and carbon dust (in the coils of the thermal compensators) so a lowering of the capacity to absorb shock loading; fatigue and creep relating to its loading and movement over the years of operation; and irradiation embrittlement which will influence its fracture mechanics performance.

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11 *Caring for Graphite Cores*, A J Wickham, CEGB Berkeley Nuclear Laboratories, 1988

12 There are a series of confidential project reports and job initiation sheets available for Sizewell A power stations which detail the corrosion of the restraint garter – methane injection was first recognised to be a problem in the late-1960s following the discovery of the extent of in-reactor steel components but work continued on the quality control of the CO_2 into the 1980s with a 4 year trial being planned for Sizewell from December 1980 – see Job Initiation 7563, December 1980. From about 1996 methane injection was reapplied at Oldbury (and possibly Wylfa) in order to reduce the rate of graphite corrosion.

Oxidation (rusting) of the restraint garter provided reason for considerable concern to the Magnox operators as late as the 1980s, with a programme then underway to determine the extent of the garter deterioration, as reported:[13]

> *"In the late 1960's this situation (corrosion inspection) was dramatically changed through an appreciation of the severe oxidation phenomena which attacks mild steel in a hot, pressurised CO_2 environment. . . .*
>
> *As a result of the initial analysis of reactor design several components were determined to be vulnerable to the effects of steel oxidation but the most important was assessed to be the Core Restraint System. The function of the restraint system is to prevent distortion of the reactor core so that satisfactory control rod and fuel element movements remain possible under varying operating conditions and, in the case of the worst credible fault of a burst CO_2 duct, that the emergency shut-down control rods can still freely enter the core and shut the reactor down. . . ."*

In fact, the oxidation programme was considered at the highest of political levels much earlier in 1970, as Cabinet Papers reveal:[14]

> *"23 November*
>
> *As you know, it was found that certain steel components inside Magnox reactors were corroding . . . at an unexpectedly high rate. The allowances made for corrosion in the design proved insufficient, and failures of inaccessible bolted components have occurred. . . . .*
>
> *21 December*
>
> *While it is too early to be certain about future prospects, the restrictions of operating temperature have markedly reduced the rate of corrosion and the CEGB expect that the design lives can be attained. Nevertheless, corrosion could make it necessary to close down some of the stations prematurely. . . .*
>
> *In the case of Wylfa even before the reactors become radioactive, it was calculated that there was no economic case for dismantling the reactors and replacing the materials susceptible to corrosion, as against operating at the restricted output throughout its life. This was mainly because of the very heavy additional cost of generation, using other stations for the several years the work would have taken.*
>
> *. . ."*

13 *Inspection Techniques at Hunterston*, T A Battle, Reactor Inspection Symposium, BNES September 1980.

14 2001 releases from the Public Records Office PREM15/134 correspondence between the Ministry for Industry R Williams and P L Gregson, 23 November, 1 December and 22 December 1970 – the concern relates to a not yet available report on the extent of corrosion by a Professor Morrison. The Morrison report itself seems to have completely disappeared for no copy has been found and made available by the libraries of the Cabinet Office, Treasury, Department of Trade, NII and Magnox Berkeley.

An internal Cabinet Office memorandum notes the safety issues involved:

> *". . .*
>
> *1 December*
>
> *The amount of corrosion is now being held more or less steady, but one or two reactors have reached a stage where another year or so's additional corrosion would make it necessary to shut them down unless a fool-proof shut-down devices recommended in the "Morrison Report" are fitted.*
>
> *22 December (to the Prime Minister)*
>
> *When you considered notes from Sir Burke Trend's Office and Sir John Eden's office on the problem of corrosion in Magnox reactors, at the beginning of this month, you asked whether, as there appears to be a continuing loss of output, it might not pay to plan a closing down and replacement programme. . . .*
>
> *Professor Morrison and I have endorsed the judgement of the Inspector of Nuclear Installations that, with regard to the effects of corrosion, the Magnox stations can be operated safely at the present time, although a further examination will be necessary during next summer's shut-downs. We have also recommended that independent nuclear shut-down devices, the functioning of which cannot be jeopardised by corrosion in the reactors, should be designed and fitted to all reactors in question. This recommendation has been accepted. Work on the devices is now going forward energetically and it is hoped that they may be ready for the two most seriously corroded reactors (Dungeness 'A' and Bradwell) by next summer.*
>
> *. . ."*

The independent nuclear shut-down devices referred to were the boron ball/dust injection systems that were fitted to all Magnox reactors, including Wylfa, during the early 1970s. The other modifications implemented to reduce the rate of corrosion included a reduction of the gas outlet temperature, effectively derating the reactors by about 15%, and, importantly, removing the methane trace content of the coolant. This latter modification resulted in the trade off between slowing the steel corrosion rate (initiated by the methane) against increasing the rate of radiolytic oxidation of the graphite (which is inhibited by the presence of methane).

Bombardment or irradiation of graphite by fast neutrons directly results in displacement of graphite atoms within the lattice structure and indirectly by gamma irradiation in radiolytic oxidation of the graphite. The lattice displacement results in a number of changes that bring about in an increased friability of the graphite; dimensional changes in both volumetric and creep with the associated material stress because the planar properties of the extruded graphite bricks are asymmetric; loss of thermal conductivity; and storage of energy. Radiolytic corrosion results in an overall weight loss and carbon dust deposition throughout the primary coolant circuit, although the rate of oxidation may be slowed by the introduction of methane and carbon monoxide.[15]

Significant amongst these age-related degradations are the reduction of the heat capacity of the moderator; the increase of stored (Wigner) energy available for subsequent release, particularly in the cooler sections of the core and blanket regimes[16] when subject to fault conditions; and the increased reactivity of graphite in air (burning) which is enhanced by contaminants entrained in the graphite pores acting as catalysts.

Fault Conditions Challenging the Core and Restraint Garter

Maintenance of the core geometry is critical under certain fault conditions. Two such fault conditions, which for the original design were considered not to challenge the core and its restraint system, are as follows:-

A) Rapid Depressurisation of the Reactor Pressure Vessel

This fault relates to failure of some part of the reactor pressure vessel containment boundary in the 'hot box' area above the core. Candidate failure localities include groups of standpipes at the pile cap, the ducting leading to the automatic pressure relief or dump valves, the ducting leading to the iodine stripping plant, and groups of services penetrations that pass through the wall of the pressure vessel

The result is a rapid depressurisation of the gas above the reactor core, a surge of gas flow through the fuel and control rod channels and the accompanying rise in gas pressure drop across the core, and the resulting upward and outward bursting force over the core. If the core restraint garter fails this results in a movement and/or distortion of the core and loss of alignment of the control rod channels with the feeding standpipes and, depending on the severity of the distortion, loss of coplanitory of the channel over its height.

15 To inhibit radiolytic corrosion a mix of about 5% carbon monoxide and 0.1% of organic material (usually methane) was added to the carbon dioxide coolant. Under irradiation the methane and products of carbon dioxide radiolysis combine to form a species that is deposited on to the pore surface of the graphite to be sacrificially oxidised. The role of the carbon monoxide is to reduce the flux of oxidising ions to the graphite pore surface, which is reducing the distance travelled by the reactive species before deactivation. Carbon deposition also arises from radiolysis of any carbon monoxide present in the coolant.

16 The fuel core of the Magnox reactor is surrounded by blanket, reflector and shielding sections of graphite – these sections run at lower graphite temperatures and are subject to neutron bombardment.

The Trawsfynydd Emergency Plan acknowledges this type of failure in setting out actions to key personnel in the immediate aftermath of a primary circuit breach:[17]

> *"If a reactor shut-down or trip has occurred, check that all safety, coarse and sector rods are fully inserted . . . some rods may be lodged in fully or partially withdrawn positions if the core structure has been disturbed as a result of a plant fault . . "*

Also, should the restraint garter fail then there is risk that channel flows would block, leading to localised overheating of fuel elements and, in an extreme core distortion or collapse, fuel elements may have been sheared by differential movement of the graphite block layers of the core.

The interbrick flow passages serve the important fault condition function of providing coolant gas to any starved fuel channel – about 15 to 20% of the total gas flow makes up the interbrick flow. Of course, interbrick flows will only establish when there is a pressure differential from one channel to another but, for a containment boundary failure above the core (ie several standpipe failing), once that the reactor circuit has fully depressurised circulation is weak and pressure differential low.

In fact there is some doubt about the effectiveness of interbrick flows in this fault condition because of the presence of carbonaceous dust created by the radiolysis and carbon monoxide processes previously discussed. This is because the accumulating dust has a tendency to block the narrow interbrick flow passages giving rise to elevated temperatures that could result in a channel fire with, first, ignition of the magnesium alloy cladding in carbon dioxide at about 700°C or, if air was present in the channel magnesium ignition at about 600°C, followed by uranium ignition of exposed uranium metal in air at about 212°C and ignition of the graphite.

Again, this channel fire ignition scenario[18] is confirmed by the Trawsfynydd action list:-

> *"Assess temperature of the core and inform the Shift Charge Engineer of the seriousness of the situation. Commence plotting the maximum CGO [Channel Gas Outlet] and Graphite temperatures as a function of time. A large graphite temperature transient occurs after a burst duct accident. The magnitude of the transient depends on the chemical reactivity [burning] of the graphite with free oxygen, the amount of oxygen (i.e. air) in the coolant and the rate at which heat can be removed from the core . . .*

17 Trawsfynydd is a steel reactor pressure vessel Magnox, although the core restraint system and the role of the graphite core is much the same as at Wylfa.

18 A channel fire occurred at Chapelcross during the late 1960s, although this related to localised heating in a channel and not, it is believed, to carbon dust accumulation.

> *. . . .Priority for action following a burst duct is first to establish an adequate gas flow through the reactor and second to adequately cool the resulting hot gas before returning it to the reactor . . .*
>
> *. . . Check the magnesium oxide and carbon monoxide sampling results from Chemical Services. If the results indicate the presence of abnormally large amounts of either compound in the coolant, it will be assumed that a channel fire has occurred . . . "*
>
> *[my additional clarification]*

These Trawsfynydd actions apply to an accident situation where it is assumed that recirculation of the coolant gas flow can be resumed by using the remaining five steam generator limbs.[19]

B) Multiple Boiler Tube Failure

During a routine inspection of one of Wylfa's reactor in April 2000 weld cracks were discovered at some of the thirty two locations where the steam superheater tails pass into the reactor pressure vessel penetration. As a result the second reactor at Wylfa was closed and both reactors have remained closed since that time.

The operator, BNFL Magnox, is currently developing a strategy to address this problem. For the longer term it is examining the feasibility of carrying out repairs to the welds concerned, subject to ALARP considerations. In the interim, in an attempt to return the reactors to service while the weld repair studies are underway, it is planning a programme of modifications to ensure that in the event of a failure of a superheater header penetration closure weld, the consequences will be acceptable.

Although details of the actual cracking are not available it is probable that the weld cracking is on the restraint plate that anchors the six or so superheater tailpipes as these are gathered together to pass into the reactor pressure vessel wall penetration. The potential failure scenario for this defect is whereby the restraint fails and the superheater tailpipes physically displace, triggering a simultaneous failure of a number of tailpipes, ejection of superheated steam from the high-pressure boiler circuit and ingress into the reactor pressure vessel.[20]

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[19] In fact, Trawsfynydd identifies another peril in that the Actions note that *"Establish an initial gas flow . . . by switching all serviceable on low speed pony motors when the circulator rotors have slowed below 600rpm. Do NOT use high speed pony motors because they are untested and their insulation may catch fire, possibly wrecking the main motor windings.".*

[20] Steam flows into the reactor from the boiler because the steamside circuit operates at a higher pressure than the reactor coolant gas (~56 bar over ~28 bar).

The worse case scenario considered in the design basis accident reactor safety case is for an abrupt and complete[21] (guillotine) failure of a single superheater tailpipe, because this fault is considered to lead to the largest credible ingress of water/steam. The escaping steam raises the reactor pressure which triggers an over-pressure trip or shut-down of the reactor and the pressure is reduced by the automatic opening of safety relief valves that vent the steam/carbon dioxide in the primary circuit to atmosphere. The situation is recovered by automatic isolation the feedwater supplies to the affected boiler with the other boiler remaining in operation for reactor decay heat removal.

In such an event, a single tube failure can result in 1 to 2 tonnes of water entering the reactor pressure vessel before the damaged boiler has been isolated from feedwater supplies.

In the event of a failure of a restraint plate[22] could result in *knock-on* damage to several superheater tailpipes. In this event, the water/steam ingress into the reactor pressure vessel could be beyond the *design basis* capability of the automatic relief valves and both the reactor pressure vessel and core restraint garter could be subjected to forces beyond the design level.[23]

The mechanism of such an abrupt injection of superheated steam into the reactor containment is quite complex. Not only is there an immediate overpressurisation of the reactor containment but, also, standing shock waves or fronts may develop as the steam 'chokes' in the confined flow areas of the fuel and control rod channels. It is these shock fronts that could generate forces within and about the core sufficient to permanently misalign the core channels or, indeed, burst the core.

Potential outcomes of such an event include:

i) distortion of the moderator core, as for the loss of containment boundary but most probably contained within a secure pressure vessel;

ii) and/or part failure of the RPV; or

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21 For part failures the moisture levels in the carbon dioxide coolant are detected as a 'leak before break' and the particular boiler is isolated.

22 At this time the Nuclear Installations Inspectorate requires further information on the condition of the welds at the superheater tailpipe liner connection with the RPV insulation and both reactors at Wylfa are shut down pending the outcome of these further investigations. In establishing these scenarios it has been necessary to make a number of assumptions on the present condition of the Wylfa reactors – this information is not available from either the nuclear industry or its regulator, the Nuclear Installations Inspectorate. That said, the assumptions made follow the rationale adopted for engineering and materials science practices and these should, in both trend and general prediction, be sound.

23 **TABLE 2** gives the working, design and test pressures for the various types of Magnox pressure vessel together with the steam side HP level – in all cases, the steamside pressure exceeds the test pressure of the RPV.

iii) catastrophic failure of the RPV

iv) failure of service penetrations to all four quadrants, thus potentially disabling all post shutdown cooling plant and forced gas circulation.

All of these possible outcomes would be accompanied by a short release of steam and coolant gas, radioactive with activation products such as sulphur-35 until the automatic safety release valves close down. Part failure of the pressure vessel might involve failure of the standpipe plugs on the reactor refuelling floor or the blow out of a service penetration reaching through the core.[24] Catastrophic failure of the RPV would be accompanied by air entrainment into the reactor and risk of a fuel (uranium) fire and a very significant release of fission products to the atmosphere.

THE ROLE OF WIGNER ENERGY[25] IN FAULT CONDITIONS

The data presented in **FIGURE 10** shows the how the rate of energy release varies with the temperature at which the graphite was initially irradiated.

On the graph, the four curves relate to identical samples of graphite that have been initially irradiated at the annotated temperatures (150, 200 ^{o}C etc). A temperature point along the bottom axis represents the temperature that might exist during a fault condition so, for each initial irradiation temperature, the rate of energy release is given by the vertical axis.

Applied to a high temperature fault condition in, say, one of the Wylfa reactors the total amount of stored energy would be about 750 MW and, with the fault temperature in excess of 400^{o}C the rate of release would be 40 MW per hour, so the duration of the release would be approximately 20 hours.[26]

Practically, the importance of stored energy in the graphite moderator cores during normal operation of the reactor has diminished due to the increased operating temperatures. However, design improvements and past temperature derating

24 Failure of a services penetration could give rise a further escalation of the event – the Wylfa reinforced concrete RPV is water-cooled so breaching the water-cooling system could result in flooding of the reactor.

25 Wigner energy in irradiated graphite can be simply considered as follows: When a neutron hits a carbon atom in the graphite lattice, it pushes the carbon atom out of position into an available interstitial space – the carbon atom is not stable in this position and Wigner energy is the potential energy it has from being out of position. When this displaced carbon atom has enough thermal energy, which is when it is hot enough, it is able to return to its original position in the lattice. Since the energy required to initiate the return to position is less than the stored or potential energy, the excess is released as heat.

26 For this calculation assume that one-third of the Wylfa core operates in the range 200 to 250^{o}C under normal conditions so a graphite mass of 3,800/3 tonnes has accumulated a maximum level of store energy of 2.1.10^{3} joules per gram, thus the total store energy is (2.1E3 x 3800/3 x E6=) 2.66.E12 J or (2.66.E12/3.6.E6=) 749MW. For the rate of release, take the average rate (**FIGURE 13**) between the bottom one-third to be 0.75 J/g^{o}C, so the release rate is (0.75 x 1.4.E12 x(400-250)=) 39.9MW/hr, so the duration of the release will be (749/39.9=) 19 hours.

result in the lower portion of the fuel core, together with the blanket and reflector sections of the core, being held at temperatures at which energy release rates can be significant.

This particularly applies to Magnox reactor plant when under fault conditions, during which (as for Scenario A) coolant flows may be impaired. In these circumstances the additional increment of Wigner energy may result magnesium clad and fuel ignition temperatures being reached.[27] Wigner energy release alone will also require a long post incident management time, extending up to 20 hours following the initiating event.

In Summary: The presence of cracking of the superheater closure welds at Wylfa is disturbing in that, first, the defect is present to some extent in all sixty four localities across both reactors and, second, failure of a single closure could result in unacceptable consequences in that this would present a *beyond design basis* event.

Because the consequences of a single closure weld failure could trigger failure of the weakened core restraint system and distortion of the graphite core, BNFL Magnox's strategy of returning the reactors to power with an interim fix[28] whilst the closure weld studies are underway, should be considered unacceptable because it continues to rely upon the integrity of the core restraint and core assembly system which, for the aged reactors cannot be stated with certainty.

[27] As well as the decay heat and Wigner energy release, if the reactor primary circuit is open to air, then the very chemically reactive carbon deposits derived from the radiolytic polymerisation of carbon monoxide in the coolant may add to the heat extraction requirement because these deposits could lead to a rapid combustion in the air ingress scenario. Also, exposure of the graphite over many years to the coolant gas carries with it the possibility of contamination of the exposed graphite surfaces by catalytic dusts that can significantly increase the reaction rates with air.

[28] The interim fix would most probably comprise a movement restraint being applied on the outer wall of the reactor pressure vessel at each penetration which would serve to resist damaging movement to the superheater tailpipes should the a penetration closure weld fail. This strategy would not address the cracking of the closure weld.

APPENDIX 1

TABLE A - MAGNOX REACTOR OPERATING CONDITIONS

PARAMETER	WYLFA
Excess Reactivity	Temp 1.8% Xe/Sm 2.13%
Max Excess Reactivity	5.52%
Control Rod Worth	7.51%
Average Fuel Burn-Up	Average 5,600MWd/tU (footnote 29)
Fuel Load	593 tU
No of Fuel Channels	6,156
No of Fuel Elements	49,248
Fuel Cladding Temp	450°C
CO2 Inlet/Outlet Temp	250/402°C
Coolant Mass tonnes	230 t CO_2
Coolant Pressure atmospheres	27.1/27.6 b
Coolant Mass Flow	10,254 kg/sec
Steam Circuit	hp only 52 b 400°C
Moderator	3,800 t Grade A Graphite
RPV Protection	11 valves
RPV Design	Working 27.1 Test 34.2
RPV Construction	29.2m dia 3.3m thick rc

DETAILS AND OPERATION OF THE WYLFA REACTORS

Magnox reactors are graphite moderated, gas-cooled reactors fuelled with elemental metal uranium at a natural enrichment level (~0.7% U^{235}).[30]

Referring to FIGURE 1, at Wylfa carbon dioxide gas is circulated within the reactor pressure vessel, through the graphite core, over the nuclear fuel and through the boilers or steam generators. The maximum circuit pressure is about 27.6 bar[31] with the gas circulators making up a circuit pressure loss of approximately 0.5 bar.

Cool gas (~250°C) is delivered to the underside or diagrid of the graphite core, it passes over and is heated (~400°C) by fuel rods located in vertical channels running through the core, and then passes to boilers transferring heat to the secondary steam circuit.

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29 This burn-up gives a fuel in-core period of four to five years. Fuel removal is before complete exhaustion of the U-235 (or at the point a beyond which the fuel can no longer sustain criticality) but at the formation of a porous annulus in the fuel rod because the reactivity in air of this loose crystal material is very high (x1000 greater than the uranium base metal) and hence it would present severe problems during an unloading incident in which the fuel was damaged.

30 It is believed that a number of Magnox reactors have been operated with slightly enriched fuel in recent years, probably to compensate for graphite moderator losses.

31 1 bar = 10.12 MN/m^2 = 1 atmosphere of pressure (14.5 pounds per square inch)

At low rates of neutron absorption, the graphite core serves to moderate the fast neutrons liberated by fission, thereby increasing the probability of subsequent fission that enables the Magnox to maintain criticality of a chain reaction of natural uranium fuel.[32]

4) MAGNOX NUCLEAR FUEL

FIGURE 2 shows a typical Magnox fuel element.

The fuel comprises a cast rod of elemental uranium (metal) alloyed with a trace of aluminium to improve its machineability at the fuel fabrication stage.

The rod is inserted and sealed within a cast magnesium oxide can (hence 'MagnOx') which is gas charged and sealed. The surface herringbone finning facilitates heat transfer and the lugs serve to locate the fuel element within the centre of a graphite core channel. Each channel will receive stack of 8 fuel elements.

Original fuel burn-ups were low at an average of 3,000 MWday/tU[33] although subsequent development of the fuel and temperature derating of the reactor plant has resulted in extended peak fuel burn-ups of up to 7,000MWday/tU.

The magnesium alloy cladding will ignite in air at ~600oC and, similarly, ignite in carbon dioxide at ~700°C. Self-ignition temperature for the uranium metal fuel is 212°C in air or at lower temperature if hydride has formed on the exposed surface.[34]

The worst-case scenario for a radioactive release to atmosphere is whereby the Magnox fuel cladding is mechanically damaged or becomes sufficiently heated to spontaneously ignite, and/or in fault circumstances whereby the carbon dioxide coolant is lost and replaced by air and the fuel metal itself ignites. Fuel ignition under open reactor circuit conditions, would result in a very significant release of radioactive fuel and fission product particles.

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32 The nucleus of the fissionable uranium-235 atom is bound together by very strong subatomic forces, so a great deal of energy is stored within an intact U^{235} atom. This energy can be released, much of it as useful heat, if the atom can be split or fissioned and rendered unstable. The heat liberated by this fission process is used to raise steam to drive the turbines that generate the electricity in a nuclear power station.

To improve the probability of a successful fission, the fast neutrons have to be slowed or 'moderated' and this is the role of the graphite core in a Magnox reactor. Graphite is chosen because its lattice structure is a very effective moderator and in doing so it absorbs few neutrons (compared to than water which is also used as a moderator in pressurised and boiling water reactors but which absorbs a larger number of neutrons, hence one of the reasons why these types of water moderated reactors require for enriched fuel). Essentially, when an atom is fissioned it breaks into two unstable fragments that immediately commence to radioactively decay. Because fission can occur in a large number of different ways, this results in hundreds of different fission products being generated within the body or matrix of the fuel. The fission products remain in the irradiated fuel, sealed in by the fuel casing or cladding.

33 MWday/tU – Megawatts days per tonne of uranium – 'burn-up' is the amount of energy liberated but usually expressed in terms of electrical output and not thermal output, which is approximately 3x larger.

34 Compounds formed the union of hydrogen with other elements, salt-like and crystalline - *Corrosion of Magnox Cladding*, Evidence to House of Commons Environment Committee, November 1985, Large & Associates, by order of the H of C Environment Committee

5) REACTOR STRUCTURE

FIGURES 3, 4, 5 and 6 show the core and pressure vessel of one of the two identical reactors at Wylfa, including the following essential components.

Graphite Moderator Core: The reactor moderator core is a stack of graphite bricks and loose keys, sitting on a framework base or diagrid. The core assembly is restrained by a radial garter system.

The integrity of the core assembly is absolutely vital for all stages of operation of the reactor. This particularly applies to the reactor cores at Wylfa because of the accelerated radiolysis weight loss sustained under the higher pressure-temperature regime of these reactors.[35] Movement and misalignment of the control rod channels during an on power incident could bar entry of the control rods, thereby negating the primary means of reactor shut-down.[36] Collapse of the core could result in fuel channel blockages and overheating of fuel elements, and/or mechanical damage (breaking) of the fuel rods.

Interbrick and Secondary Flows: FIGURE 7 shows how the individual graphite bricks are arranged to form the fuel and control rod channels vertically through the height of the core. Notches are formed in the top and bottom faces of bricks of the intermediate layers with the across-core passages created providing for 'interbrick' flow of coolant gas through the core from channel to channel. Interbrick flow compensates for any small pressure differentials from channel to channel and, more generally, contributes to flattening temperature profile across the core.

Interbrick flow provides fuel cooling in the event of a fuel channel blockage. This is because downstream of the blockage the channel pressure drops thus drawing in greater rates and volumes of interbrick flow from the higher pressures of the adjacent channels.

In the Wylfa design (FIGURE 5) the interbrick flow is sourced from a secondary flow of coolant gas at the reactor inlet temperature. This secondary or re-entrant flow is routed up the annular gap between the core and the shield or reactor 'tank' wall, thence via the interbrick passages across the graphite core.

Core Restraint Garter: FIGURE 6 is a schematic of the garter, comprising series of pivoted and tilting beams, radial restraints, interconnected by steel bands with thermal compensators acting around the periphery of the core, serves to restrain movement and contain the outward thrust from within the core. The garter reacts against movements and forces within the core generated by temperature changes and the pressure drop (about 0.3b) of the gas flowing through core channels. Over the operational lifetime of the reactor, the garter has to compensate for volumetric changes of the graphite due to radiolytic weight loss.

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35 *Radiolytic Graphite Oxidation*, Progress in Nuclear Energy, 1985, 16, 127-178, A J Wickham, CEGB Nuclear Laboratories.

The restraint garter is a crucial safety element for a number of loss of coolant and pressure transient fault conditions, where the core internals may be subject to very rapid and high-pressure differentials. Failure of the garter during these types of fault condition would enable the fault train to cascade to a serious accident scenario.

Fuel and Control Rod Standpipes: Access for fuel charging and control rod operation is via standpipes that run from the charge face in the refuelling hall, through the 'lid' of the reactor pressure vessel. Each fuel standpipe provides for access to a cluster of 16 adjacent channels, with the standpipes pitched at approximately 0.8m centres - FIGURE 7.

The refuelling and control rod geometry will tolerate a small degree of lateral shift (~50mm) of the core before alignment is lost, although this margin progressively decreases with age because of dimensional changes (shrinkage) of the graphite brought about by irradiation.

Each fuel standpipe is sealed at the reactor charge face (the floor of the refuelling hall) with a plug that is accessed by the fuel charging machine.[37, 38] The control rod standpipes terminate in a pit housing the control rod motor.

Both control and fuel standpipes each individually form part of the pressure vessel containment boundary. The closure plugs of all standpipes have to withstand high pressure transients during certain fault conditions.

Reactor Pressure Vessel (RPV): The reinforced concrete pressure vessels at Wylfa are not prone to irradiation embrittlement and its pre-stressing steel tendons are sufficiently shielded from neutron irradiation (although irradiation embrittlement applies the reactor inner steel components, such as the core restraint garter, core tank and certain areas of the pressure vessel liner).

Potential age related degradation factors for the reinforced concrete pressure vessels include thermal cycling of the concrete, cyclic creep of the prestressed tendons, and carbonation of the concrete surface, particularly at the services penetrations.[39] Concrete will crack and spall when subject to high temperatures, so the gas-tight steel liner protects the inner surface of the RPV with a thermal insulation backing.

Each reactor pressure vessel is the primary containment boundary, being the single and final barrier between a reactor fault condition that involves fuel cladding damage and the release of fission product radioactivity into the atmosphere.

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37 Other than at Hunterston A where the fuel was removed from the bottom of the reactor.

38 *The standpipes at Hinkley Point A sustained corrosion and required in-situ repairs – Standpipe Distortion at Hinkley Point A Power Station and the Cost of Decommissioning Magnox Reactors*, 2nd Report from the Energy Committee, 1986-87 – see also, Memorandum *Standpipe Distortion/Thinning at Hinkley Point A and Decommissioning Costs*, Large & Associates 1986

39 Carbonation is the infusion of carbon dioxide into the surface of the concrete whereby it gives rise to the formation of micro-cracking and fissuring and which may result in concrete spalling in localised areas, particularly where the services penetration reach through the RPV walls – the inner steel liner of the RPV prevents direct contact with the carbon dioxide coolant.

The reactor safety design (which gives rise to the Design Basis Accident) pivots around a damage severity that is related to the particular type of Magnox reactor. For example, the maximum tolerable breach area of the primary circuit for a steel, spherical RPV of the earlier Magnox steel RPV reactors is taken to be the abrupt failure of the lower gas duct of about 1 m diameter (0.8 m^2). For the reinforced concrete pressure vessel design adopted for Wylfa the maximum tolerable breach area is likely to be much smaller at 0.03m^2 for an ex-quadrant breach and 0.006m^2 in-quadrant,[40] or matched to the most vulnerable services penetration or standpipe grouping.

The in-quadrant breach particularly applies to service penetrations such as where the bundled boiler water feed and superheater steam tails pass to and from the boilers through the concrete walls of the pressure vessel – as schematically represented by FIGURE 7. The locality of the welded junction between the steel insulation plating and the penetration liner tube, on the inside of the RPV, not only provides the opportunity for a in-quadrant breach but also for disruption of the low pressure, water system channelled inside the RPV walls that serves to cool the RPV concrete in the locality of the service penetrations.

It is believed that this is the location of the present suspect welds at the Oldbury and Wylfa reactors and the main reason why both reactors at Wylfa are presently shut-down. A requirement to provide features to facilitate the repair of such a defect was not included within the original design and this may explain why the reactors at Wylfa have been shut-down for such a long period. Incidentally, if it is a generic defect then it may also apply to the Advanced Gas-Cooled Reactors that deploy similar cooled service penetrations through the reinforced concrete RPV.

6) FUEL AND RPV CONTAINMENT

The Magnox design provides a single level of containment beyond the fuel cladding. At Wylfa this single containment comprises the reinforced concrete pressure vessel, which is generally considered so massive as to be failsafe, the standpipes, the ducting that leads to the pressure relief valves and each of the services and the water and steam pipe penetrations that pass through the pressure vessel walls.

The magnesium alloy fuel cladding encapsulates the fuel rod and fission products generated during irradiation. During the process of irradiation the uranium metal fuel rods swells and tends to bond with the cladding and the fuel, unlike uranium dioxide fuel pellets it is not susceptible to cracking. Thus the fuel-clad gap inventory is relatively small, although there is a tendency for the transuranic products to migrate to the fuel rod boundaries.

[40] The breach areas are those applied to the AGR rc pressure vessel but most probably apply to Wylfa because of the similarity of the design. The later reactor safety systems assume that the reactor plant (boilers, circulators, etc) is formed into four 'quadrants' each with its own independent services area – steam and CO^2 penetrations from within the RPV pass into these quadrant areas. An 'in-quadrant' fault is where the breach delivers the escaping CO_2, steam etc., into a single quadrant support area (manned motor and switch rooms, etc) thereby disabling its systems but with at least two other quadrants remaining fully and independently operational. An ex-quadrant fault is where the coolant escapes elsewhere and does not affect the operation of any one quadrant's equipment.

In a fault condition where the primary circuit has been breached and the fuel cladding damaged, and irrespective of the immediate post-fault aftermath conditions, the release of radioactivity to the environment will most certainly include almost all of the gaseous and volatile fission products that accumulate in the 'clad gap'. Thereafter, the thermo-chemical-mechanical conditions that develop in the aftermath determine the fraction of the fission products held in the matrix of the fuel that will release to the environment – but see footnote 29.

The concrete structure of the pressure vessel also serves as a biological shield and support structure for the reactor refuelling floor. There is no secondary containment to act as a failsafe should the reactor primary circuit breach under fault conditions and there is nothing in the Magnox fuel cladding design that includes for additional stability, robustness, or whatever, in compensation for this lack of secondary containment.

The safety reasoning for Wylfa is that a rapid depressurisation, that is failure of the concrete pressure vessel, is a *beyond design basis* fault – this type of fault condition could result in distortion of the graphite core, misalignment of the control rod channels and difficulty in inserting the control rods to shut the reactor down. However, since it was assumed that the core would remain stable under all design basis fault conditions, Wylfa was not originally fitted with diverse means of shutdown.[41]

7) BOILERS

The boilers are housed within the reactor pressure vessel and are of the 'once-through' type of design that, essentially, dispenses with the intermediate header drums and any recirculation of the fluid. In this type of boiler, each of the multitude of thin-walled tubes passes completely through the unit from pre-heater to superheater stages without entering header drums. The tube material is changed to suit each of the pre-heat, boiling and superheating processes underway (low chromium for preheat, stress corrosion resistant high chromium for boiling and creep resistant austenitic for superheating).

Since there is no distinctive separation of the processes underway in any single tube, a change in the fluid conditions within the tube results in that particular process relocating up or down the tube – these undesirable movements are compensated for by changing either the temperature or pressure conditions acting in the boiler. For example, if there is a sudden drop in the steamside pressure then boiling will commence earlier, further down in the tube – if the boiling regime relocates in the low-chromium section of the tube then rapid failure may occur due to stress corrosion of the tube material. Similarly, if the gas side temperature suddenly drops, then the boiling regime moves up the tube where it may quench the austenitic superheater tubing and promote brittle failure. Obviously, in a multi-tube boiler the same conditions apply across the whole bank of tubes, so loss of

[41] Following the Generic Issues studies, in about 1995 Wylfa was fitted with articulated control rods to account for core distortion as a result of a seismic event. The earlier Magnox stations which do cater for core distortion under a rapid depressurisation event due to a burst duct failure, include a secondary shut down system whereby boron balls or beads are dropped into the reactor channels to terminate the nuclear reaction and the later AGR reactors include nitrogen purge system in which high pressure nitrogen gas floods the reactor thereby suppressing nuclear activity. It is believed that a boron dust injection system is fitted at Wylfa.

control over the processes, particularly under certain reactor fault conditions, could result in multiple tube failure.

The final collection of the individual tubes at the end of the superheater section can also provide opportunity of simultaneous tube failures. In the once-through boiler design at Wylfa[42] the individual superheater tubes are collected together via a series of sleeved subheaders - this is an arrangement where two tubes run into one larger tube, with two of these being run into a larger tube still, and so on until a few number of superheater tailpipes are bundled together in a steel tube liner which passes through the RPV wall. Failure of a single superheater tailpipe represents the bounding limit of the design basis accident, although failure of a cluster superheater tailpipes is equivalent to a multiple tube failure and beyond the design basis.

Since the steam side (secondary) of the boilers operate at significantly higher pressure than the carbon dioxide coolant gas in the reactor primary circuit, any boiler tube failure will result in rapid ingress of water/steam into the reactor, immediately thereafter a degree of cooling of the reactor pressure vessel accompanied by a rapid rise in reactor pressure. The extent of these temperature and pressure variations will be dictated, in terms of temperature, by the location of the boiler failure (lower down then a greater thermal cooling shock) and, in the magnitude of the pressure transient, by the number of tubes that simultaneously fail.

8) PRIMARY SAFETY SYSTEMS

For nuclear reactivity control, the earlier Magnox steel rector pressure vessel power stations include control rod insertion and boron bead (or balls) injection as diverse means of emergency shutdown. Wylfa power station is not fitted with the boron ball diverse means of shut-down, nor nitrogen purge which is deployed in the later AGR reactors, although it has a boron dust injection system and a number of articulated control rods which are designed to operate should the graphite core laterally shift during a fault condition.

The rate of loss of primary circuit gas pressure, once detected and at a predetermined threshold will initiate an automatic reactor trip. Similarly, the rate of boiler water ingress into the reactor will be detected by moisture content transducers located at the output of each boiler so that boiler can be identified and isolated. If the boiler tube failure is rapid and precedes the so-called ‘leak-before-break’ detection systems, then primary circuit overpressure sensors trip the reactor and the automatic overpressurisation valves open to dump the (radioactive) coolant directly to atmosphere.

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42 The precise details of the boiler systems at Wylfa are not publicly available, most probably of the once-*through* type because these boilers were prototype designs for the following AGR nuclear power stations. Being early prototypes, the steam superheaters may terminate in headers or sub-headers.